3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 3, "Design of Structures, Systems and Components" of the Tennessee Valley Authority's (TVA's or the applicant's) Construction Permit Application (CPA), Preliminary Safety Analysis Report (PSAR). TVA submitted the CPA for a small modular reactor (SMR) at the Clinch River Nuclear (CRN) Site located in Oak Ridge, Roane County, Tennessee. The PSAR is based on the proposed construction of a one-unit BWRX-300 SMR (hereinafter referred to as CRN-1) designed by GE-Vernova Hitachi Nuclear Energy (GVH) with a nominal electrical output of 300 MWe. The staff's regulatory findings documented in this SER are based on Revision 1 of the TVA CPA PSAR, dated April 9, 2026 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML26119A628).

The staff reviewed Chapter 3 of the CRN-1 PSAR, as supplemented, against applicable regulatory requirements using regulatory guidance and standards to assess the sufficiency of the preliminary information on Design of Structures, Systems and Components for the issuance of a construction permit (CP) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 (TN249), "Domestic Licensing of Production and Utilization Facilities." As part of this review, the staff evaluated information on Design of Structures, Systems and Components, with special attention given to design and operating characteristics, unusual or novel design features, and principal safety considerations. The staff evaluated the preliminary design of the Design of Structures, Systems and Components to ensure the design criteria, design bases, and information relative to construction are sufficient to provide reasonable assurance that the final design will conform to the design basis. The NRC staff's reviews and evaluations of subject matter relevant to CRN-1 PSAR Chapter 3, including regulations and guidance used, a summary of the application information reviewed, and evaluation findings and conclusions, are discussed in the safety evaluation (SE) sections below.

3.1 Compliance with NRC General Design Criteria

Section 3.1 of the TVA's PSAR, "Compliance with U.S. Nuclear Regulatory Commission General Design Criteria," contains an evaluation of the GVH boiling water reactor (BWR), BWRX-300 standard plant design, analyses, procurement, and construction requirements for compliance with 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

The BWRX-300 design incorporates passive safety features that are designed to safely shut down the reactor and maintain it in a safe shutdown condition without reliance on alternating current power, external cooling water, or operator action for anticipated operational occurrences (AOOs) and design-basis accident (DBA) postulated initiating events (PIEs). The motive power for the passive safety systems is supplied by stored energy sources such as direct current power, compressed gas, springs, or gravity. In addition, the BWRX-300 design is provided with a complement of highly reliable systems that reduce the challenge to the passive safety systems and provide the capability for extending the safe shutdown period.

Because the general design criteria (GDC) in 10 CFR Part 50, Appendix A are not technology neutral and are predicated on legacy design, the applicant substituted some GDC with principal design criteria (PDC), specifically PDC 17, 19, 26, 28, and 29, in accordance with 10 CFR 50.34(a)(3)(i) and (ii). Although Appendix A, "Advanced Reactor Design Criteria," to Regulatory

Guide (RG) 1.232 (NRC 2018-TN7066), *Guidance for Developing Principal Design Criteria for Non-Light-Water-Reactors*, was developed for six non-light-water reactor technologies, the applicant used the guidance to inform the development of the BWRX-300 PDC.

Table 3.1-1 in the applicant's PSAR provides references to the other PSAR sections where additional design and analysis information is discussed. GDC 1–5 are omitted from Table 3.1-1 in the PSAR because they have plant-wide applicability. The staff's review and assessment of how the applicant addressed the plant-specific GDC and PDC are documented in the relevant chapters of this safety evaluation report. Section 3.1 of the PSAR also includes a writeup of the applicable GDC statements and evaluation against the GDC on meeting the applicable requirements of 10 CFR 50, Appendix A.

3.2 Classification of Structures, Systems, and Components

PSAR Section 3.2, "Classification of Structures, Systems, and Components," references licensing topical report NEDC-33934P, "BWRX-300 Safety Strategy," Revision 2, that describes the approach to classifying structures, systems, and components (SSCs) as identifying functions that affect nuclear safety, assigning a safety category to those functions based on their importance, and then assigning a safety class to the components that perform those functions.

The primary factors influencing the outcome of SSC classification are (1) consequences of failure to perform the functions, (2) expected frequency of an SSC being called on to perform its functions, and (3) time following a postulated initiating event (PIE) at which, or the period for which, the SSC performs its functions.

Safety importance of plant functions (i.e., defense line [DL] functions) is determined based on how functions are credited to mitigate PIEs and event sequences in the deterministic safety analysis. The organization of PIEs and event sequences is based on event categories, which reflect the frequency of occurrence of the PIE or event sequence from the probabilistic safety assessment (PSA).

Licensing topical report NEDC-33934P, Revision 2 defines the fundamental safety functions (FSFs) that protect physical barriers. For a given event sequence, if the functional DLs required to fulfill the FSFs are performed successfully, then the corresponding physical barriers remain effective. The FSFs for the BWRX-300 are (1) control of reactivity, (2) removal of heat from the fuel, and (3) confinement of radioactive material, shielding against radiation, and control of planned radiative release, as well as limitation of accidental radioactive releases.

The evaluation of (1) assignment of safety category to functions, (2) assignment of safety class to components, (3), seismic categorization, and (4) quality group is provided below.

3.2.1 Assignment of Safety Category to Functions

3.2.1.1 Introduction

This section of the PSAR references licensing topical report NEDC-33934P, "BWRX-300 Safety Strategy," Revision 2, which provides a description of the defense lines, safety class, safety categories, and associated mapping to NRC regulatory requirements and guidance. The BWRX-300 uses a defense in depth framework that employs two types of defensive layering: (1) physical barriers to radiological release; and (2) active, passive, and inherent safety functions to protect these barriers. Safety functions are assigned to specific defense lines (DLs) based on their safety importance; these defense line functions are then assigned to Safety Categories (i.e., Safety Category 1, 2, 3, or N). In this framework, DL3 functions are considered the most important, followed by DL4a functions (which backup DL3 functions). DL2 and DL4b functions are considered less important than DL3 and DL4a functions, but they provide a means to address postulated initiating events and are considered important to safety. Consistent with this approach NEDC-33934P, Table 4-1, "Functional Safety Category Assignment," provides the following mapping between DL functions and Safety Categories:

- DL3 functions are assigned to Safety Category 1;
- DL4a functions are assigned to Safety Category 2; and
- DL2 and 4b functions are assigned to Safety Category 3.

Functions not assigned to Safety Category 1,2, or 3 are assigned to Safety Category N.

3.2.1.2 Regulatory Evaluation

10 CFR Part 50.34(a)(3) (TN249) requires the PSAR to include the preliminary design for the facility, including identification of principal design criteria for the facility. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," establishes the minimum for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission. 10 CFR 50.34(a)(4) requires the PSAR to include the preliminary analysis and evaluation of the design and performance of SSCs of the facility with the objective of assessing the risk to public health and safety. Consistent with 10 CFR 50, Appendix A, GDC 1, "Quality standards and records," SSCs important to safety are be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Appropriate safety classification ensures that appropriate quality standards are applied to identified safety functions.

3.2.1.3 Technical Evaluation

The NRC's regulations do not use the term "safety category"; however, assignment to a safety category directly links to the BWRX-300 safety class as described in Section 3.2.2 of this report. Therefore, the staff reviewed the safety category assignment process to verify that the process would support appropriate assignment of safety class consistent with NRC requirements. Specifically, components required to perform DL functions are assigned to the following Safety Categories:

- Safety Category 1: DL3 primary and integral support functions required within the first 72 hours of an event are classified as Safety Category 1. As discussed in NEDC-33934P, Table 2-1, "Overview of BWRX-300 Defense Lines," DL3 functions detect and mitigate design basis accidents (DBAs).

- Safety Category 2: DL3 functions needed between 72 hours and 7 days following an event as well as DL4a functions needed within the first 7 days of an event are classified as Safety Category 2. As noted in NEDC-334934P, Table 2-1, DL4a functions detect and mitigate design extension category events and some DBAs associated with failure of DL3 functions

- Safety Category 3: DL3 and DL4a functions needed after 7 days as well as primary and integral support functions for DL2 and DL4b. As noted in NEDC-334934P, Table 2-1, DL2 functions detect and mitigate anticipated operational occurrences (AOOs) while DL4b functions detect and mitigate DECs to prevent core damage or to mitigate core damage events.

Safety Category 1 covers functions that are safety-related (10 CFR 50.2) while Safety Categories 2 and 3 generally cover important to safety (e.g., 10 CFR 50, Appendix A) functions.

3.2.1.4 Conclusion

The NRC staff finds that PSAR and NEDC-334394P adequately describe the safety classification framework for mapping safety functions to safety categories. However, the regulations in 10 CFR 50 do not use the term "safety categories." The relationship between safety categories and safety class, which is mapped to NRC regulatory terminology, is discussed in Section 3.2.2 of this report.

3.2.2 Assignment of Safety Class to Components

3.2.2.1 Introduction

This section references NEDC-33934P, Revision 2 for the description of the process for assigning safety class to SSCs based on the safety category supported by SSCs. Specifically, NEDC-33934P, Table 5-1, "Safety Class and Category Mapping to NRC Regulatory Requirements and Guidance," maps safety categories to the associated safety class. Safety classes are then linked to NRC terminology used in regulations and guidance.

3.2.2.2 Regulatory Evaluation

10 CFR Part 50.34(a)(3) (TN249) requires the PSAR to include the preliminary design for the facility, including identification of principal design criteria for the facility. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," establishes the minimum for the principal design criteria for watercooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission. 10 CFR 50.34(a)(4) requires the PSAR to include the preliminary analysis and evaluation of the design and performance of SSCs of the facility with the objective of assessing the risk to public health and safety. Consistent with 10 CFR 50, Appendix A, GDC 1, "Quality standards and records," SSCs important to safety are to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Appropriate safety classification ensures that appropriate quality standards are applied to identified safety functions.

3.2.2.3 Technical Evaluation

NEDC-33934P, Table 5-1 provides the following mapping between Safety Category, Safety Class, and NRC terminology:

- Safety Class 1 (SC1): SC1 includes Safety Category 1 functions. In addition, SC1 also includes components that make up the fission product barriers, components that are part of the RCPB, and components that are part of design provisions that perform an FSF, whose failure is considered “practically eliminated.” SC1 SSCs are equivalent to “safety related” SSCs as defined in 10 CFR 50.2

- Safety Class 2 (SC2): SC2 includes Safety Category 2 functions and SSCs supporting Regulatory Treatment of Non-Safety Systems (RTNSS) functions.
- Safety Class 3 (SC3): SC3 includes Safety Category 3 functions and SSCs supporting Regulatory Treatment of Non-Safety Systems (RTNSS) functions.

SC2 and SC3 are equivalent to nonsafety-related, but important to safety, SSCs (e.g., those nonsafety-related SSCs needed to meet the principal design criteria). The applicant notes that there is not a one-to-one correlation between the RTNSS scoping criteria and the SC2 and SC3 definitions, therefore RTNSS can be classified into either of these SCs. Non-Safety Class (SCN) is assigned to all other components classified by NRC as non-safety related, including those that perform make-ready support functions for DL2, DL 4b, and normal functions.

Table 3A-1, “Preliminary Classification of Structures, Systems, and Components,” in the PSAR identifies the safety classification of components for the BWRX-300 reactor. The staff’s review of the preliminary safety classification of these SSCs is discussed in the associated section of this report.

3.2.2.4 Conclusion

The NRC staff finds that the classification framework described in PSAR Section 3.2.1 and 3.2.2 and NEDC-33934P provides an adequate description of the process used is determine SSC safety class assignments. Specifically, the assignment of SSCs meeting the definition of safety-related and non-safety related but important to safety is acceptable because it is consistent with the applicable definitions in 10 CFR 50.2 and 10 CFR 50, Appendix A, Therefore, staff finds that the preliminary safety class assignments provided in the PSAR meet the regulatory requirements of 10 CFR 50.34(a) (3) and (4), and adequately support the issuance of a CP pursuant to the regulations in 10 CFR 50.35 and 10 CFR 50, Appendix A.

3.2.3 Seismic Categorization

3.2.3.1 Introduction

This section refers to NEDC-33934P, Revision 2 for the seismic categorization of SSCs and summarizes the categorization using nomenclature selected for the CRN-1 Site.

3.2.3.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), “Preliminary safety analysis report,” sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for seismic classification of SSCs.
- In GDC 1 and Appendix B to 10 CFR Part 50, the NRC requires applying QA provisions to activities affecting the safety functions of SSCs designed as seismic Category I commensurate with their importance to safety.
- In GDC 2, the NRC requires, in part, that SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

- In GDC 60, the NRC requires the nuclear power unit design to include means to control suitably the release of radioactive materials in gaseous and liquid effluents.
- In Appendix A to 10 CFR Part 100 (TN282) and Appendix S to 10 CFR Part 50, the NRC requires certain SSCs be designed to withstand the safe shutdown earthquake (SSE) and remain functional.

NUREG-0800 (NRC 2021-TN8013), *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition* (SRP), Section 3.2.1, "Seismic Classification," lists acceptance criteria adequate to meet the above requirements. The following NRC regulatory guides (RGs) contain relevant guidance for this review:

- RG 1.29, *Seismic Design Classification for Nuclear Power Plants*, describes an acceptable method of identifying and classifying those plant features that must be designed to withstand the effects of the SSE (e.g., seismic Category I features) (NRC 2021-TN12804).
- RG 1.151, *Instrument Sensing Lines*, provides guidance with regard to seismic design provisions and classification of safety-related instrument sensing lines (NRC 2020-TN13171).
- RG 1.143, *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants*, identifies several radioactive waste SSCs requiring some level of seismic design consideration (NRC 2001-TN1134.
- RG 1.189, *Fire Protection for Nuclear Power Plants*, provides guidance used to establish the design provisions of fire protection SSCs, including seismic design consideration and seismic classification for certain SSCs (NRC 2023-TN12814).

3.2.3.3 Technical Evaluation

PSAR Section 3.2.3, "Seismic Categorization," states that seismic categorization of structures and components is based on the importance of functions performed, as described in NEDC-33934P, Revision 2. The seismic categorization approach described in NEDC-33934P, Revision 2, is an alternative approach to RG 1.29. The BWRX-300 seismic categorization approach is based on the importance of functions being performed within the BWRX-300 design. NEDC-33934P, Revision 2, Section 4.5, "Seismic Categorization" classifies structures and components for management and storage of radiological material to the criteria in RG 1.143.

PSAR Appendix 3A, "Preliminary Classification of Structures, Systems, and Components" identifies seismic categorization of components. Because this information is not required for issuance of a CP, the NRC staff will review the details of the component level seismic classification during the operating license application process when the design details are available.

PSAR Appendix 3A references NEDC-33934P, "BWRX-300 Safety Strategy," Revision 2. Note 9 to Table 3A-1, "Preliminary BWRX-300 Component Classification List," states that components classified as SC1 may be assigned to a seismic category lower than I. GVH has proposed PSAR markup to revise Note 9 to Table 3A-1 to be consistent with NEDC-33934P, Revision 2 in response to Audit Question A-3A-1. The NRC staff has reviewed the audit response and finds the proposed final safety analysis report (FSAR) change acceptable as the seismic categorization is consistent with NEDC-33934P, Revision 2.

As discussed above and in the evaluation for NEDC-33934P, Revision 2, the seismic classification of the SSCs is consistent with the guidance in SRP 3.2.1 and RG 1.143 and provides an acceptable alternative approach to RG 1.29, therefore, also meets the requirements in GDC 1, 2, and 60; 10 CFR Part 50, Appendix B; and 10 CFR Part 100, Appendix A.

Conclusion

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the applicant's description of the plan to classify SSCs important to safety and implement the QA Program meet the regulatory requirements of 10 CFR 50.34(a)(3)(ii), and meets provisions of RG 1.29, RG 1.143, and SRP 3.2.1; therefore, the information provided by the applicant meets the relevant requirements of GDC 1, 2, and 60; 10 CFR Part 50, Appendix B; and 10 CFR Part 100, Appendix A, and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.2.4 Quality Group

3.2.4.1 Introduction

PSAR Section 3.2.4 designates the use of RG 1.26, Quality Group Classifications and Standards for:

- Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, for establishing the appropriate codes and standards based on the importance of the pressure-retaining function of BWRX-300 components.

3.2.4.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for designating quality group for SSCs.
- 10 CFR Part 50, Appendix A, GDC 1, and 10 CFR 50.55a, as they relate to SSCs important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.
- 10 CFR 50.55a(c)(1), as it relates to components that are part of the reactor coolant pressure boundary (RCPB) that must meet the requirements for Class 1 components in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, except as provided in 10 CFR 50.55a(c)(2) through (4).
- 10 CFR 50.55a(d)(1), as it relates to components classified as Quality Group B that must meet the requirements for Class 2 components in ASME BPVC, Section III.
- 10 CFR 50.55a(e)(1), as it relates to Quality Group C components that must meet the requirements for Class 3 components in ASME BPVC, Section III.

SRP Section 3.2.1, "Seismic Classification," lists acceptance criteria adequate to meet all the above requirements. The following NRC RG contains relevant guidance for this review.

RG 1.26, Revision 6 (NRC 2021-TN12803) describes an acceptable method for determining quality standards for Quality Groups B, C, and D water- and steam-containing components important to the safety of water-cooled nuclear power plants.

3.2.4.3 Technical Evaluation

PSAR Section 3.2.4, "Quality Group," states that quality group designation per RG 1.26 is used for establishing the appropriate codes and standards based on the importance of the pressure-retaining function of BWRX-300 components. Components are classified as Quality Group A, B, C, D, or as not applicable. The NRC staff finds that the quality group classification of SSCs is consistent with RG 1.26 and therefore acceptable to meet the requirements of GDC 1, 10 CFR 50.34(a)(3)(ii), and 10 CFR 50.55a listed above. PSAR Table 3A-1 identifies the quality group of components. The NRC staff will review the details of the component level quality group classification during the operating license application process when the design details are available.

3.2.4.4 Conclusion

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the applicant's description of the plan to classify pressure-retaining components of fluid systems important to safety as Quality Group A, B, C, or D items meet the regulatory requirements of 10 CFR 50.34(a)(3)(ii), and meets the requirements of GDC 1 and 10 CFR 50.55a through the use of RG 1.26. Therefore, the staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements 10 CFR 50.34(a) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.2.5 Conclusion

As discussed above, the NRC staff finds that the provisions for assignments of safety category and safety class, seismic classification, and quality group meet the NRC relevant regulations.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

3.3.1.1 Introduction

PSAR Section 3.3.1, "Wind Loadings," describes the design severe wind speed and recurrence interval, its variation with height, and the applicable gust factors that define the input parameters to account for wind loading for structural design. PSAR Section 3.3.1 also summarizes the procedures used for transforming the design wind speed to effective wind pressure and distribution in determining wind loading forces applied to the structures. Relevant consensus standard ASCE/SEI 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," is identified along with relevant NRC guidance.

3.3.1.2 Regulatory Evaluation

10 CFR 50.34(a)(3)(i) (TN249), requires a construction permit application to include principal design criteria that meet 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," which establishes the minimum requirements for principal design criteria for

water-cooled nuclear power plants. The staff evaluated the applicant's compliance with the following NRC regulations, which contain the relevant requirements for this review:

- 10 CFR 50.34(a), "Preliminary safety analysis report," sub-paragraph (3)(ii) requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for protection against severe wind natural phenomena.
- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as severe wind and extreme wind without loss of capability to perform their safety functions, with consideration of (1) the most severe weather phenomena historically reported for the site and surrounding area and (2) appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena, and (3) the importance of the safety functions to be performed.
- SRP Section 3.3.1, Revision 3, "Wind Loading," issued March 2007, lists acceptance criteria adequate to meet the above requirements and provides review interfaces with SRP Sections 2.3.1 and 2.3.2.
- ASCE/SEI 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," American Society of Civil Engineers, 2016.

3.3.1.3 Technical Evaluation

The staff reviewed PSAR Section 3.3.1 against the regulatory requirements identified above and the relevant acceptance criteria in SRP Section 3.3.1.

2.3.1, "Regional Climatology," and 2.3.2, "Local Meteorology," of this report document the staff's evaluations of the adequacy of the most severe regional and local meteorological data used to specify design wind load parameters.

The staff noted that the site-specific value of basic wind speed (96.3 mph 3-second gust) in PSAR Table 2.0-1R, which is from the early site permit application and calculated using ASCE/SEI 7-05 with a recurrence interval of 100 years, remains the design-basis wind speed for the CRN-1 site. The staff further noted that the CRN-1 site-related design parameter used for the design of seismic Category 1 structures is a basic 3-second gust wind speed of 71.5 m/s (160 mph) and corresponds to a mean recurrence interval of 3,000 years in accordance with the more recent ASCE/SEI standard 7-16 wind maps for Risk Category IV (essential facilities). The staff finds that both the design-basis wind speed as well as the design parameter wind speed (160/1.15 = 139 mph when adjusted for ASCE/SEI 7-05 importance factor that is implicitly incorporated in the ASCE 7-16 risk category-based wind maps) bound the historically reported maximum wind speed of 87 mph (3-second gust) per PSAR Table 2.0-1R. The staff also noted that the design wind speeds are translated to equivalent pressure and applied forces using procedures in ASCE/SEI 7-16, which the staff reviewed as a later edition of the ASCE/SEI 7-05 standard referenced in SRP 3.3.1, and finds acceptable as a recognized and proven updated consensus standard for determination of wind loads on structures, which bounds the ASCE 7-05 design procedures and parameters.

The staff finds that the applicant has adequately defined the design bases and design parameters for severe wind to be applied to the preliminary design of important-to-safety plant structures consistent with the importance of safety function (risk category IV) performed and with sufficient margin to meet GDC 2 and prevent structural damage during the most severe wind loadings determined to be appropriate for the site. The staff also finds the applicant has specified methods provided in ASCE/SEI 7-16, which the staff reviewed and found acceptable, to transform wind speed into equivalent pressure and to select pressure coefficients corresponding to the structure's geometry and physical configuration to determine applied wind forces on the structures. The procedures used to determine design wind loadings on structures are acceptable because they are in accordance with ASCE/SEI 7-16, which is a recognized consensus standard for determining design wind loads on structures that the staff finds acceptable consistent with the design parameters as defined therein. Load combinations for the design of seismic Category 1 structures including severe wind loads in combination with appropriate loads resulting from normal and accident conditions are addressed in Section 3.8.4 of this report. Therefore, the staff finds that the design-basis parameters and methods for severe wind design meet the relevant acceptance criteria of SRP Section 3.3.1 and thus meet the requirements of GDC 2.

3.3.1.4 Conclusion

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the applicant's design-basis parameters and methods for severe wind loading meet the relevant acceptance criteria of SRP Section 3.3.1. Therefore, the severe wind design bases for the CRN-1 important-to-safety structures are adequate to meet relevant requirements of GDC 2 to demonstrate capability of the structures to withstand design severe wind loading so that their design reflects appropriate consideration for the most severe wind recorded for the site with an appropriate margin, appropriate combinations of the effects of normal and accident conditions with the effects of the severe wind natural phenomena, and the importance of the safety function to be performed. The NRC staff, thus, finds that the information provided by the applicant is sufficient for preliminary design and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35. The staff will confirm that the final design conforms to this design basis during the evaluation of the CRN-1 FSAR.

3.3.2 Extreme Wind (Tornado and Hurricane) Loadings

3.3.2.1 Introduction

PSAR Section 3.3.2 "Extreme Wind (Tornado and Hurricane) Loadings," defines the tornado and hurricane wind design input parameters applicable to seismic Category 1 structures by reference to PSAR Table 2.0-1R and states the tornado wind design parameters bound the hurricane wind parameters. PSAR Section 3.3.1 also summarizes the procedures for transforming the tornado wind speed into pressure-induced forces and distribution applied to the structures. Relevant topical report BC-TOP-3A, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," and consensus standard ASCE/SEI 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," are identified along with relevant NRC guidance.

3.3.2.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for protection against extreme wind natural phenomena.

- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of extreme wind natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety functions, with consideration of (1) the most severe natural phenomena historically reported for the site and surrounding area and (2) appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

SRP Section 3.3.2, Revision 3, "Tornado Loads," issued March 2007, lists acceptance criteria adequate to meet the above requirement and provides review interfaces with SRP Sections 2.3.1 and 2.3.2. The following NRC regulatory guides (RGs) contain relevant guidance for this review.

- RG 1.76 (NRC 2007-TN3294), *Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants*, describes acceptable design-basis tornado-generated missile spectra for the design of nuclear power plants.
- RG 1.221 (NRC 2011-TN5931), *Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants*, describes acceptable design-basis hurricane-generated missile spectra for the design of nuclear power plants.
- RG 1.143, Revision 2 (NRC 2001-TN1134), *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants*."
- NEDO-33914-A, Revision 2, "BWRX-300 Advanced Civil Construction and Design Approach," Licensing Topical Report (GE Hitachi 2022-TN13022).

3.3.2.3 Technical Evaluation

The staff reviewed PSAR Section 3.3.2 against the regulatory requirements identified above and the relevant acceptance criteria in SRP Section 3.3.2.

Sections 2.3.1, "Regional Climatology," and 2.3.2, "Local Meteorology," of this report document the staff's evaluations of the adequacy of the most severe regional and local meteorological data used to specify extreme wind (hurricane and tornado) load parameters.

Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds," of this report documents the staff's evaluations of the adequacy of the design-basis hurricane and tornado wind-generated missile spectrum.

Section 3.5.3, "Barrier Design Procedures," of this report documents the adequacy of the procedure for transforming the tornado missile impact into an equivalent static force on the impacted structure.

The design-basis hurricane and tornado design parameters for seismic Category 1 structures at CRN-1 are defined in PSAR Table 2.0-1R. From this table, the design parameter for hurricane windspeed is 200 mph, which bounds the site-specific value of 130 mph. The tornado design parameters, namely, maximum pressure drop, maximum rotational speed, maximum translational speed, maximum wind speed, radius of maximum rotational speed, and rate of pressure drop, are, respectively, 1.2 psi, 184 mph, 46 mph, 230 mph, 150 ft, and 0.5 psi/s, which are equal to the corresponding site-specific values. The staff finds the design parameters for hurricanes and tornadoes to be acceptable because they are consistent with RG 1.221 and RG 1.76, respectively, for the location of the CRN-1 site. From the design parameters listed in

Table 2.0-1R, the staff finds that the tornado wind design parameters bound the hurricane wind parameters.

The civil structures for storage and processing of radioactive waste (e.g., Radioactive Waste Building [RWB]) classified as RW-IIa class, in accordance with RG position 5 of RG 1.143 are designed to withstand the tornado wind demand of Table 2 of RG 1.143, which the staff finds acceptable because it is consistent with the guidance in RG 1.143 for such structures. The staff also finds the procedures for transforming tornado wind speed into pressure-induced forces applied to the structures and the distribution based on topical report BC-TOP-3A acceptable because the procedures are based on an NRC-approved topical report and the approach is similar to a later standard, ASCE 7-05, referenced in SRP Section 3.3.2. The tornado loading considers wind pressures, differential pressure loads due to rapid atmospheric pressure change, and tornado- or hurricane-generated missiles, which the staff finds acceptable because it is consistent with the guidance in SRP Section 3.3.2 and RG 1.76.

The seismic Category II power block structures surrounding the reactor building (RB) are designed to maintain structural integrity during an extreme wind event such that they do not collapse on the seismic Category 1 RB. The staff finds this approach acceptable because the wind interaction evaluation follows the interaction requirements and methodology in Section 6.0 of NRC-approved licensing topical report NEDO-33914-A.

The staff finds that the applicant has adequately defined the design bases and design parameters for extreme wind (tornado and hurricane) to be applied to the preliminary design of important-to-safety plant structures consistent with the importance of safety function performed and with sufficient margin to meet GDC 2 and prevent structural damage during the most severe tornado wind loadings determined to be appropriate for the site. The staff also finds the applicant has specified appropriate methods provided in BC-TOP-3-A, which is an NRC-approved topical report, to transform tornado wind speed into equivalent pressure and size coefficient to determine applied tornado wind forces on the structures. Load combinations for the design of seismic Category 1 structures including extreme wind loads in combination with appropriate loads resulting from normal and accident conditions are addressed in Section 3.8.4 of this report. Therefore, the staff finds that the design-basis parameters and methods for extreme wind (tornado and hurricane) design meet the relevant acceptance criteria of SRP Section 3.3.2 and thus meet the requirements of GDC 2.

3.3.2.4 Conclusion

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the applicant’s design-basis parameters and methods for extreme wind (tornado and hurricane) loading meet GDC 2 by meeting the relevant acceptance criteria of SRP Section 3.3.2. The NRC staff thus finds that the information provided by the applicant is sufficient for preliminary design, meets the regulatory requirements 10 CFR 50.34(a)(3)(ii) and other regulatory requirements identified in this section, and adequately supports the issuance of a CP pursuant to 10 CFR 50.35. The staff will confirm that the final design conforms to this design basis for extreme wind during the evaluation of the CRN-1 FSAR.

3.3.3 Conclusion

Based on the reviews in Sections 3.3.1 and 3.3.2 of this report, the NRC staff concludes that the applicant has provided sufficient design-basis information for severe and extreme wind (hurricane and tornado) loadings for preliminary design of CRN-1, meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other regulatory requirements identified in the sections, and adequately supports the issuance of a CP pursuant to 10 CFR 50.35. The staff will confirm that the final design conforms to this design basis for severe and extreme wind loading during the evaluation of the CRN-1 FSAR.

3.4 Water Level (Flood) Design

3.4.1 Internal Flood Protection for Onsite Equipment Failures

3.4.1.1 Introduction

This section discusses the SSCs to be protected from internal flooding, whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity in accordance with the requirements of GDC 2 and 4.

PSAR, Table 3A-1, lists all the SSCs (safety-related and nonsafety-related) in various locations of the plant (including inside and outside the containment) and identifies for each SSC the associated seismic category, quality group, and safety classifications. The building and site structure seismic category and seismic design basis is provided in Table 3A-2. Safety Class 1 (SC1) SSCs listed in Table 3A-1 are to be protected from internal flood.

3.4.1.2 Regulatory Evaluation

The applicable regulatory requirements for the evaluation of the CRN-1 SSCs for flood damage are as follows:

- 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for protection against internal flooding.
- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as flooding without loss of capability to perform their safety functions.
- GDC 4, "Environmental and dynamic effects design bases," requires, in part, that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents.

The staff reviewed the BWRX-300 design for protecting SSCs important to safety against internal flooding in accordance with Revision 3 of SRP Section 3.4.1, "Internal Flood Protection for Onsite Equipment Failures."

3.4.1.3 Technical Evaluation

The staff evaluated the sufficiency of the Clinch River preliminary design features for systems and components, as described in Clinch River PSAR Section 3.4.1, for the issuance of a CP using the guidance and acceptance criteria from SRP Section 3.4.1 with regard to flood protection as part of meeting GDC 2 and GDC 4.

The applicant provided a description of the internal flood protection and methodology for evaluation of potential flooding. **[[**

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The piping failure criteria and evaluation are described in Section 3.6 of PSAR. SSCs are also evaluated for risk effects in the internal flooding probabilistic safety assessment (PSA) using the methods described in Section 15.6.3.7.

The staff reviewed the information provided in Section 3.4.1 and finds the level of detail provided on internal flooding is adequate for the preliminary design. **[[**

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The staff will review specific SSC layouts, water volumes, termination of flow, floor drains, slopes, and mitigating features during the operating license (OL) application. **[**

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3.4.1.4 Conclusion

Based on its review and evaluation, the staff finds that the facility design features for coping with internal flooding meet the applicable guidance and regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

Further technical or design information required to complete the safety analysis may reasonably be left for later consideration. The staff will confirm that the final design conforms to GDC 2 and 4 during the evaluation of the FSAR for the OL.

3.4.2 Analysis Procedures for Protection of Structures Against Flood from Extreme Sources

3.4.2.1 Introduction

PSAR Section 3.4.2, "Analysis Procedures for Protection of Structures Against Flood from External Sources," describes **[[**

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3.4.2.2 Regulatory Evaluation

The following NRC regulation contains the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for protection against external flooding natural phenomena.
- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as floods without loss of capability to perform their safety functions, with consideration of (1) the most severe natural phenomena historically reported for the site and surrounding area and (2) appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

SRP Section 3.4.2, Revision 3, "Analysis Procedures," issued March 2007, lists acceptance criteria adequate to meet the above requirement and provides review interfaces with SRP Sections 2.4.3 and 2.4.12.

3.4.2.3 Technical Evaluation

The staff reviewed PSAR Section 3.4.2 against the regulatory requirements identified above and the relevant acceptance criteria in SRP Section 3.4.2 with regard to analysis procedures as part of meeting GDC 2.

Sections 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers," and 2.4.12, "Ground Water," of this report document the staff's evaluations of the flood and ground water site design parameters.

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Further, load combinations for the design of seismic Category 1 structures including loads from hydrostatic effects of maximum flood and groundwater levels in combination with appropriate loads resulting from normal and accident conditions are addressed in Section 3.8.4 of this report.

3.4.2.4 Conclusion

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the applicant meets the guidelines of SRP Section 3.4.2 as it relates to the design bases and analytical methods for protection of CRN-1 structures against flooding from external sources sufficient to meet GDC 2. The NRC staff, thus, finds that the information provided by the applicant is sufficient for preliminary design, meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other regulatory requirements identified in this section, and adequately supports the issuance of a CP pursuant to 10 CFR 50.35. The staff will confirm that the final design conforms to this design basis for protection against external flood sources during the evaluation of the CRN-1 FSAR.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

3.5.1.1.1 Introduction

This PSAR section contains a brief description of methods used to identify and define internally generated missiles outside containment. Section 3.5.1 identifies the methodology to determine when a missile is defined as statistically significant that requires missile protection.

As indicated in PSAR Section 3.5, Safety Class 1 (SC1) SSCs listed in Table 3A-1 are designed to be protected from the missiles. Section 3A.1 of the PSAR indicates that Table 3A-1 provides a preliminary list of the principal BWRX-300 components.

3.5.1.1.2 Regulatory Evaluation

The following NRC regulations are relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), “Preliminary safety analysis report,” sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for protection against internally generated missiles.
- GDC 4 as it relates to the design of the SSCs important to safety, to assess whether the design affords protection from the internally generated missiles that may result from equipment failure.

SRP Section 3.5.1.1, “Internally Generated Missiles (Outside Containment),” of NUREG-0800 (NRC 2021-TN8013) provides relevant regulatory guidance that all SSCs (outside containment) are to be protected from internally generated missiles to ensure compliance with 10 CFR Part 50, Appendix A, GDC 4 requirements, as well as interfaces with other relevant SRP sections.

3.5.1.1.3 Technical Evaluation

The staff reviewed the Clinch River design for protecting SSCs important to safety against internally generated missiles (outside containment) in accordance with the guidance of SRP Section 3.5.1.1.

SRP Sections 3.5.1.1, “Internally Generated Missiles (Outside Containment),” and 3.5.1.2, “Internally-Generated Missiles (Inside Containment),” delineate that SSCs important to safety are to be protected from internally generated missiles to ensure compliance with GDC 4 requirements. This includes internally generated missiles from component overspeed failures; missiles that could originate from high-energy fluid system failures; and missiles caused by, or as a consequence of, gravitational effects (e.g., falling or dropping equipment). An internally generated missile is a dynamic effect of such failures, and its impact on SSCs that are important to safety must be evaluated. Protecting SSCs from the effects of internally generated missiles ensures the capability to shut down and maintain the reactor in a shutdown condition and the capability to prevent a significant uncontrolled release of radioactivity.

The applicant summarized in PSAR Section 3.5.1.1 that an evaluation is performed for potential internally generated missiles that could result from failure of the plant equipment located outside the containment. The applicant defined potential missiles internally generated outside containment as those from rotating equipment, high-pressure system ruptures, and missiles caused by, or as a consequence of, gravitational effects (e.g., heavy load drop).

The staff reviewed the design for protection against pressurized gas bottles. In response to Audit item A-3.5-5, the licensee confirmed gas bottles are integral part of hydraulic control unit design (PSAR Section 4.6.3.2). In addition, the licensee indicated a rack of pressurized nitrogen bottles is located in the RB. Section 3.5.1.1 the PSAR (Rev 1) was updated to contain general criteria used for the BWRX-300 for protection against pressurized gas bottles, and additional information will be provided in the FSAR.

In PSAR Section 3.5.1, the applicant described the process of determining proper missile protection. Once a potential missile is identified, its statistical significance is determined by the combined probability of an event that is defined as the product of the following:

- Probability of missile occurrence (P1);
- Probability of impact on a significant target (P2);
- Probability of significant damage (P3); and,
- Combined probability (P4 = P1 × P2 × P3).

If the combined event probability of a potential missile is greater than 1×10^{-7} per year, the missile is considered statistically significant, and protection of safety-related SSCs against the missile will be provided. If the combined event probability of a potential missile is less than 1×10^{-7} per year, the missile and their consequences are not considered statistically significant, the missile is dismissed, and protection of safety-related SSCs against the potential missile would not be provided.

Once a statistically significant missile is identified, protection of SSCs against the missile will be provided as indicated in PSAR Section 3.5.1 by one or more of the following methods:

- location of the system or component in an individual missile-proof structure
- physical separation of redundant systems or components of the system from the missile trajectory path or calculated range
- provision of localized protection shields or barriers for systems or components
- design of the particular structure or component to withstand the impact of the most damaging missile

- provision of design features on the potential missile source to prevent missile generation
- orientation of the potential missile source to prevent unacceptable consequences caused by missile generation

The staff did not verify application of the methodology, nor verify statistical significance of potential missiles, as these details were not provided in PSAR. However, the staff understands, as indicated in PSAR Section 3.5, SC1 SSCs listed in Table 3A-1 are designed to be protected from the missiles and will be evaluated for protection against all statistically significant missiles.

The PSAR doesn't contain any discussion of specific missile type (i.e., pressure components, rotating equipment, valve parts, etc.) or basis to discredit any potential source of missile—for example, discrediting an actuator or bonnet due to ASME code or a missile not generated based on location within plant (i.e., no rotating equipment near containment). However, the applicant has listed options for mitigating measures in case a statistically significant missile is identified as shown above. Therefore, the staff understands that PSAR Section 3.5 statement "Safety Class 1 (SC1) SSCs listed in Table 3A-1 are designed to be protected from the missiles," as meaning that all SC1 SSCs will be analyzed for protection against all types of statistically significant missiles. Additionally, all non SC1 equipment, components, or structures will be evaluated to ensure failure could not result in a missile which causes failure of SC1 equipment.

During an audit, the applicant concurred with the staff understanding related to the missile protection approach described above. The staff review of the information provided in PSAR Section 3.5.1.1 finds the level of detail provided on methodology to address internally generated missile protection is adequate for the preliminary design.

Based on its review, the staff concludes that the applicant's approach to protect safety-related SSCs against the effects of internally generated missiles outside containment is consistent with requirements of GDC 4.

3.5.1.1.4 Conclusion

The applicant provided a description of the process to define SSCs requiring missile protection and methodology that will be used to evaluate missile protection. Based on its review, the staff finds that the facility design features for coping with internal missiles meet the applicable guidance and regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35

Further technical or design information required to complete the safety analysis may reasonably be left for later consideration. The staff will confirm that the final design conforms to this design basis during the evaluation of the Clinch River FSAR.

3.5.1.2 Internally Generated Missiles (Inside Containment)

3.5.1.2.1 Introduction

This PSAR section contains a brief description and indicates similar missiles are evaluated as discussed in Section 3.5.1.1 of the PSAR. Section 3.5.1 identifies the methodology to determine when a missile is defined as statistically significant that requires missile protection.

As indicated in PSAR Section 3.5, Safety Class 1 (SC1) SSCs listed in Table 3A-1 are designed to be protected from the missiles. Section 3A.1 of the PSAR indicates that Table 3A-1 provides a preliminary list of the principal BWRX-300 components.

3.5.1.2.2 Regulatory Evaluation

The following NRC regulations are relevant requirements for this review:

In 10 CFR 50.34(a), “Preliminary safety analysis report,” sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for protection against internally generated missiles.

- GDC 4 as it relates to the design of the SSCs important to safety, to assess whether the design affords protection from the internally generated missiles that may result from equipment failure.

SRP Section 3.5.1.2, “Internally-Generated Missiles (Inside Containment),” of NUREG-0800 (NRC 2021-TN8013) provides relevant regulatory guidance that all SSCs (inside containment) are to be protected from internally generated missiles to ensure compliance with 10 CFR Part 50, Appendix A, GDC 4 requirements as well as interfaces with other relevant SRP sections.

3.5.1.2.3 Technical Evaluation

The staff reviewed the Clinch River design for protecting SSCs important to safety against internally generated missiles (inside containment) in accordance with the guidance of SRP Section 3.5.1.2.

SRP Section 3.5.1.2, “Internally-Generated Missiles (Inside Containment),” delineates that SSCs important to safety are to be protected from internally generated missiles to ensure compliance with GDC 4 requirements. This includes all SSCs within the containment and the containment itself. This includes internally generated missiles from component overspeed failures; missiles that could originate from high-energy fluid system failures; and missiles caused by, or as a consequence of, gravitational effects (e.g., falling or dropping equipment). An internally generated missile is a dynamic effect of such failures, and its impact on SSCs that are important to safety must be evaluated. Protecting SSCs from the effects of internally generated missiles ensures the capability to shut down and maintain the reactor in a shutdown condition and the capability to prevent a significant uncontrolled release of radioactivity.

The applicant indicated in PSAR Section 3.5.1.1 that potential missile sources from both rotating equipment and pressurized components are considered, when applicable. PSAR Section 3.5.1.2 clarifies that gravitational missiles inside the containment are also considered.

In PSAR Section 3.5.1, the applicant described the process of determining proper missile protection. Once a potential missile is identified, its statistical significance is determined by the combined probability of an event that is defined as the product of the following:

- Probability of missile occurrence (P1);
- Probability of impact on a significant target (P2);
- Probability of significant damage (P3); and
- Combined probability (P4 = P1 × P2 × P3).

If the combined event probability of a potential missile is greater than 1×10^{-7} per year, the missile is considered statistically significant, and protection of safety-related SSCs against the missile will be provided. If the combined event probability of a potential missile is less than 1×10^{-7} per year, the missile and their consequences are not considered statistically significant, the missile is dismissed, and protection of safety-related SSCs against the potential missile would not be provided.

Once a statistically significant missile is identified, protection of SSCs against the missile will be provided as indicated in PSAR Section 3.5.1 by one or more of the following methods:

location of the system or component in an individual missile-proof structure

- physical separation of redundant systems or components of the system from the missile trajectory path or calculated range
- provision of localized protection shields or barriers for systems or components
- design of the particular structure or component to withstand the impact of the most damaging missile
- provision of design features on the potential missile source to prevent missile generation
- orientation of the potential missile source to prevent unacceptable consequences caused by missile generation

- The staff did not verify whether methodology has been acceptably applied, nor verify statistical significance of potential missiles, as these details were not provided in PSAR. However, the staff understands, as indicated in PSAR Section 3.5, SC1 SSCs listed in Table 3A-1 are designed to be protected from missiles and will be evaluated for protection against all statistically significant missiles.

- Based on its review, the staff concludes that the applicant’s approach is consistent with that described in SRP Section 3.5.1.2 above related to protecting safety-related SSCs against the effects of internally generated missiles inside containment and is consistent with the recommendations in SRP Section 3.5.1.2, and thus meets GDC 4.

3.5.1.2.4 Conclusion

The applicant provided a description of the process to define SSCs requiring missile protection and methodology that will be used to evaluate missile protection. Based on its review, the staff finds that the facility design features for coping with internal missiles meet the applicable guidance and regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

Further technical or design information required to complete the safety analysis may reasonably be left for later consideration. The staff will confirm that the final design conforms to this design basis during the evaluation of the Clinch River FSAR.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Introduction

GDC 4 requires SSCs important to safety to be appropriately protected against dynamic effects of postulated accidents, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. One potential source of plant missiles is the rotor of the main turbine. As such, the applicant must consider this potential source of turbine missiles in the plant's design and must protect SSCs important to safety from the adverse effects of postulated turbine missiles.

The objective of the staff's review is to determine whether the potential turbine missiles have been appropriately identified and whether the SSCs important to safety have been appropriately protected from any adverse effects that may result from these missiles.

3.5.1.3.2 Regulatory Evaluation

10 CFR 50.34(a), "Preliminary safety analysis report," sub-paragraph (3)(ii), requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for addressing the orientation, probabilistic analysis, inspection and testing criteria used to protect essential SSCs from turbine missiles, and sub-paragraph (3)(iii) requires applicant to provide information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin of safety.

GDC 4 states that SSCs important to safety shall be appropriately protected against dynamic effects, including the effects of missiles that may result from equipment failures. The steam turbine has a heavy rotor that rotates at high speeds during normal operating conditions and could generate high-energy missiles that have the potential to damage safety-related SSCs.

RG 1.70, Revision 3, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants – LWR Edition, November 1978 (NRC 1978-TN12879).

RG 1.115 provides guidance on the evaluation of the effect of turbine missiles and the issues related to the probability of turbine missile generation, which is the focus of the staff's evaluation in this section of this report (NRC 2012-TN12842).

3.5.1.3.3 Technical Evaluation

The staff reviewed and evaluated the information in Section 3.5.1.3 of the PSAR, supplemented in response to audit questions, to ensure that the evaluation of turbine missiles are in accordance with the criteria of SRP Section 3.5.1.3 and RG 1.115 to meet the requirements of GDC 4. The staff believes the applicant provided enough information in the CRN-1 PSAR in accordance with the guidance of RG 1.70 to support issuance of a CP.

3.5.1.3.3.1 *Turbine Orientation and Methodology*

The failure of a rotor in a steam turbine may result in the generation of high-energy missiles that could affect essential SSCs. These essential SSCs should be adequately protected from the effects of turbine missiles such that functions important to safety are maintained. RG 1.115 provides three methodology approaches for protecting essential SSCs: (1) favorable orientation of the turbine unit such that all essential SSCs are outside the missile strike zone, (2) limiting the frequency of turbine missile generation, or (3) use of barriers to protect essential SSCs.

The applicant selected to use the favorable orientation methodology per PSAR Section 3.5.1.3, which states that the turbine is favorably oriented and all important to safety SSCs are outside of the low trajectory strike zone. There are no descriptions of the location of the SSCs to be protected from turbine missiles, and therefore, the determination that the turbine generator orientation is favorable cannot be made. Audit question 3.5-8 requested the SSCs to be protected and the applicant provided an adequate response to this question (see audit summary report at NRC 2026-TN13251). Further information on turbine orientation can reasonably be left for later consideration in the final safety analysis report (FSAR) since this information is not necessary to be provided as part of a CPA.

3.5.1.3.3.2 *Turbine Missile Probability Analysis*

Per PSAR Section 3.5.1.3, the turbine vendor has not been selected, and the type of turbine will be a 3,600 rpm turbine, in lieu of a 1,800 rpm turbine. Per RG 1.115, the turbine missile strike zone of 25 degrees is for an 1,800 rpm turbine, and therefore, the strike zone for the 3,600 rpm and the associated P1, P2, and P3 probabilities will be modified to accommodate the 3,600 rpm turbine. Further information on turbine probability analysis can reasonably be left for later consideration in the FSAR since this information is not necessary to be provided as part of a CPA.

Based on the above, the turbine missiles will be evaluated during the OL review once a turbine has been selected and actual SSCs and locations are determined and built. The review will consist of verifying that the as-built locations of the important to safety SSCs are not within the modified strike zone and verifying the modified P1, P2 and P3 probabilities for the 3,600 rpm turbine. The detailed turbine missile analysis will be evaluated based on these modified probabilities in order to determine the inspection of the turbine rotor and testing frequencies of the turbine control systems (including valves).

3.5.1.3.4 Conclusion

The staff determined that the applicant provided sufficient information on the turbine in accordance with guidance in RG 1.70 and meets the regulatory requirements of 10 CFR 50.34(a) and that the detailed review of the turbine missile analysis will be reviewed during the operating license application once the turbine has been selected and the as built locations of the important to safety SSCs are established. Based on the above, the staff will complete its review at the operating license stage to make a determination with respect to GDC 4 per RG 1.115, that the specific criteria for orientation, probabilistic analysis, inspection and testing programs will protect essential SSCs from turbine missiles. Therefore, the NRC staff concludes that the information provided by the applicant with respect to following the guidance of RG 1.115 for protecting essential SSCs from turbine missiles is acceptable to support the issuance of a CP pursuant to the NRC regulations in 10 CFR 50.35, with specific details to be reviewed during the operating license stage.

3.5.1.4 Missiles Generated by Tornadoes and Extreme Winds

3.5.1.4.1 Introduction

This section identifies and describes missiles generated by extreme winds (such as a tornado or hurricane). PSAR Section 3.5.1.4, "Missiles Generated by Extreme Winds," describes the spectrum of missiles generated by extreme winds that are applicable to the Clinch River Plant, which includes massive high-kinetic-energy missiles, rigid missiles, and small rigid missiles. PSAR Table 3.5-1 contains the missile spectrum applicable to protection and design of Reactor Building, Main Control Room, and Main Control Room to Secondary Control Room Egress.

3.5.1.4.2 Regulatory Evaluation

The following NRC regulations are relevant requirements for this review:

In 10 CFR 50.34(a), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for protection from missiles generated by extreme winds.

- In GDC 2, the NRC requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety functions.
- In GDC 4, the NRC requires, in part, that SSCs important to safety shall be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit.

SRP Section 3.5.1.4, "Missiles Generated by Tornadoes and Extreme Winds," provides relevant regulatory guidance. The following NRC regulatory guides (RGs) contain relevant guidance for this review.

- RG 1.76, *Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants*, describes acceptable design-basis tornado-generated missile spectra for the design of nuclear power plants (NRC 2007-TN3294).
- RG 1.221, *Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants*, describes acceptable design-basis hurricane-generated missile spectra for the design of nuclear power plants (NRC 2011-TN5931).

3.5.1.4.3 Technical Evaluation

The staff reviewed the Clinch River design for protecting SSCs important to safety against missiles generated by extreme winds in accordance with the guidance of SRP Section 3.5.1.4. The applicant indicated the methodology used for designing BWRX-300 complies with RG 1.76 for tornado missiles, RG 1.221 for hurricane missiles, and RG 1.143 missile spectrum for the RWB with modifications to reconcile with more recent guidance in ANSI/ANS-2.3, "Estimating Tornado, Hurricane, and Extreme Straight Line Wind Characteristics at Nuclear Facility Sites," consistent with RG 1.76.

PSAR Section 3.5.1.4 describes design-basis tornado and hurricane winds and associated missile spectra for the Clinch River design as follows:

design-basis extreme wind parameters (PSAR Table 2.0-1R)

A tornado has a maximum wind speed of 230 mph.

A hurricane has a design-basis wind speed of 200 mph (bounds site-specific value of 130 mph for a 3-second gust noted in RG 1.221)

tornado-generated missile spectra

A massive high-kinetic-energy missile that deforms on impact, which is defined as 1,810 kg (4,000 lb) automobile with dimensions of 5 m by 2 m by 1.3 m (16.4 ft by 6.6 ft by 4.3 ft), has a horizontal velocity of 41 m/s (135 feet per second (ft/s)) and a vertical velocity of 27.5 m/s (90.5 ft/s).

A rigid missile that tests penetration resistance, which is defined as a Schedule 40 pipe being 0.168 m diameter × 4.58 m long (6.625 in diameter × 15 ft long), 130 kg (287 lb), has a horizontal velocity of 41 m/s (135 ft/s) and a vertical velocity of 27.5 m/s (90.5 ft/s).

A small rigid missile of a size that is sufficient to pass through openings in protective barriers, such as a 0.0669 kg (0.147 lb), 2.54 cm (1 in) diameter solid steel sphere, has a horizontal velocity of 8 m/s (26 ft/s) and a vertical velocity of 5.4 m/s (17.6 ft/s).

hurricane-generated missile spectra

A massive high-kinetic-energy missile that deforms on impact, such as the automobile described above, has a horizontal velocity of 56 m/s (183 ft/s), based on windspeed of 89.4 m/s (200 mph).

A rigid missile that tests penetration resistance, such as the Schedule 40 pipe described above, has a horizontal velocity of 43.5 m/s (142 ft/s).

A small rigid missile of a size that is sufficient to pass through openings in protective barriers, such as the solid steel sphere described above, has a horizontal velocity of 38 m/s (125 ft/s).

The design-basis vertical missile velocity for all hurricane missiles is 26 m/s (85.3 ft/s).

The applicant has assumed that the automobile missiles will impact at all heights up to 9.14 m (30 ft) above grade within 0.5 miles (0.8 kilometers) of the plant structures. Pipe and sphere are evaluated at full height of the building. The staff reviewed the above information and finds it acceptable because applying the automobile missile only to elevations below 9 m (30 ft) is consistent with the guidance of RG 1.76 and RG 1.221.

The guidance of RG 1.76 applies only to the continental United States, which is divided into three regions: Region I, the central portion of the United States; Region II, a large region of the United States along the east coast, the northern border, and western Great Plains; and Region III, the western United States. The tornado parameter values specified in RG 1.76, Table 1, for Region I are most severe and bound all the tornado parameter values specified for

Regions II and III. The staff finds that the above design-basis tornado parameters provided by the applicant and tornado-generated missile spectra are conservatively in accordance with the guidance in RG 1.76, Table 1, for Region I. RG 1.221 provides contour maps of the U.S. coastal areas most susceptible to hurricanes and associated design-basis wind and missile speeds. The staff finds that the above design-basis hurricane parameters and hurricane-generated missile spectra for the Clinch River design are in accordance with the guidance in RG 1.221.

Section 2.3 of this report contains the staff's evaluation of the meteorological site parameters. The staff evaluates the structural performance of the Clinch River design with respect to hurricane and tornado missiles in Section 3.8 of this report.

Table 3.5-1 of the PSAR also contains wind-generated missile spectrum applicable to the RWB, which is consistent with RG 1.143 guidance and further evaluated in Section 3.3.2 and 11.2.

Based on its review, the staff finds that the information provided by the applicant conforms to the guidance in RG 1.76 and RG 1.221 for design-basis tornado and hurricane missiles, respectively. Therefore, the staff concludes that the Clinch River design meets the requirements of GDC 2 and GDC 4, with respect to the protection of SSCs important to safety from natural phenomena such as tornadoes and hurricanes.

3.5.1.4.4 Conclusion

The staff's review concludes that the applicant's design-basis tornado and hurricane-generated missile spectra for the Clinch River design comply with the requirements in 10 CFR Part 50, Appendix A; GDC 2; and GDC 4 for SSCs to be protected from missiles generated by extreme winds because the applicant meets the acceptance criteria in SRP Section 3.5.1.4 and sufficiently defined missiles in accordance with RG 1.76, RG 1.221, and RG 1.143 for design-basis windborne missiles for nuclear power plants.

Based on above, the NRC staff finds that the information in PSAR Section 3.5.1.4 is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(1)(ii) and other regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

PSAR Section 3.5.1.5 on Site Proximity Missiles (Except Aircraft) points to Section 2.2.3.2 of the PSAR, which evaluated potential accidents that could generate missiles (i.e., explosions). Staff's evaluation of PSAR Section 2.2.3.2 is in Section 2.2.3 of this report and it concluded there are no offsite activities with the potential for missile generation.

3.5.1.6 Aircraft Hazards

In PSAR Section 3.5.1.6, TVA evaluated the core damage frequency (CDF) from an aircraft crash because the total aircraft crash frequency, as described in PSAR Section 2.2.2.7.4, is greater than 1E-07 per year. The staff review of the annual aircraft hazards at the proposed facility is presented in Section 2.2.1.1.2 of this report. As reviewed in Section 2.2.1.1.2 of this report, general aviation aircraft transiting the nearby airways contributes the majority of the estimated annual hazard.

In PSAR Section 3.5.1.6, TVA estimated the CDF for the Turbine Building (TB), Control Building (CB), Radwaste Building (RWB), and Service Building (SB) from crash of a small general aviation aircraft. In addition, the application stated in PSAR Appendix 3A that the Reactor Building (RB), which houses the reactor, is able to withstand a crash of a large commercial aircraft without perforation. The NRC staff evaluated this in Appendix 3N of this report and found that it agreed with that assertion. Thus, all safety class I (SC-1) components within the RB would maintain their intended safety functions after a large commercial aircraft has crashed onto it. Therefore, all SC-1 components would maintain their intended safety functions if a small general aviation aircraft crashes because the engines and landing gear, the most rigid components of an aircraft that cause the maximum damage to a structure in an aircraft crash, have significantly smaller mass. Therefore, for those reasons, the staff agrees that the reactor building, which contains most of the radionuclides, would be able to withstand a crash of a general aviation aircraft and the CDF, as estimated in the PSAR would be acceptably small.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

3.5.2.1 Introduction

This section discusses the SSCs to be protected from externally generated missiles, including the approach for protection of all safety-related SSCs required to safely shut down the reactor and maintain it in a safe condition.

The applicant defined the SSCs to be protected from externally generated missiles as those that are required to safely shut down the reactor and maintain it in a safe condition and indicated SSCs are typically protected by the structure containing them. Seismic Category I and RW-IIa structures, and portions of the control building (CB) structure are designed to withstand the effects of externally generated missiles.

3.5.2.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), “Preliminary safety analysis report,” sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for protection against externally generated missiles.
- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety functions.
- GDC 4 requires, in part, that SSCs important to safety shall be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit.

SRP Section 3.5.2, “Structures, Systems and Components to be Protected from Externally-Generated Missiles,” provides relevant regulatory guidance. The following NRC regulatory guides (RGs) also contain relevant guidance for this review:

- RG 1.13, *Spent Fuel Storage Facility Design Basis*, Revision 2, as it relates to the capacity of the spent fuel pool cooling systems and structures to withstand the effects of externally generated missiles and to prevent missiles from contacting the stored fuel assemblies (NRC 2007-TN13096)

- RG 1.27, *Ultimate Heat Sink for Nuclear Plants*, Revision 2, as it relates to the capability of the ultimate heat sink and connecting conduits to withstand the effects of externally generated missiles (NRC 1976-TN13102)
- RG 1.115, as it relates to the protection of the SSCs important to safety from the effects of turbine missiles (NRC 2012-TN12842)
- RG 1.117, as it relates to the protection of the SSCs important to safety from the effects of tornado missiles (NRC 2016-TN13104)

3.5.2.3 Technical Evaluation

The staff evaluated the sufficiency of the Clinch River preliminary design features for systems and components, as described in Clinch River PSAR Section 3.5.2, for the issuance of a CP using the guidance and compliance with GDC 2 and GDC 4. The staff review was based on acceptance criteria from Section 3.5.2, "Structures, Systems, and Components to be Protected from Externally-Generated Missiles," of NUREG-0800, Section 3.5.2 (SRP).

The applicant considers extreme wind (tornado and hurricane) generated missiles as the bounding externally generated missile on a plant site. PSAR, Table 3A-1, lists all the SSCs (safety-related and non-safety-related) in various locations of the plant and identifies the associated seismic category, quality group, and safety classifications. As indicated in PSAR Section 3.5.2, SSCs that are required to be protected from external missiles are located in buildings that are designed to be tornado resistant. Seismic Category I and RW-IIa structures, and portions of the CB structure, defined by the different systems requirements, are designed to withstand the effects of externally generated missiles. Further review of SSCs and buildings requiring protection, and acceptability of resistant classification will be evaluated in OL stage.

The applicant indicated in PSAR Section 3.5.2 that the design for missiles generated by extreme winds conforms with RG 1.76 and RG 1.221, as discussed further in Section 3.5.1.4 of this report. The applicant indicates conformance with RG 1.117 and RG 1.13 related to SSCs to be protected from externally generated missiles, which is further defined in SRP Section 3.5.2. The applicant further states that RWB missile protection is in accordance with RG 1.143; this is further evaluated in Section 3.3.2 and 11.2 of this report for radioactive systems. Additionally, PSAR Table 1.9-3 indicates conformance with RG 1.27 and alternate conformance with RG 1.115, which is reviewed in Section 3.5.1.3 of this report.

The staff reviewed GDC 2 conformance based on adherence to RGs 1.76 and 1.221 related to extreme wind missiles. As indicated in PSAR Section 3.5.2, design basis missile protection conforms with methods defined in RGs 1.76, 1.221, and 1.117. Missiles are defined in PSAR Section 3.5.1.4 and protection of safety-related SSCs are located within protective buildings. Additionally, the PSAR indicates that the main control room (MCR) and MCR to secondary control room egress route are hardened to meet RG 1.117 criteria to provide a qualified route for egress.

The ultimate heat sink (UHS) design described in PSAR is not a typical UHS as addressed in RG 1.27 for cooling the reactor facilities for operating plants but must be protected from externally generated missiles. The heat sink for the plant consists of a normal heat sink (NHS) and UHS, which are defined in Sections 9.2.5 and 9.2.9 of PSAR. The UHS provides the safety-related heat sink to mitigate DBAs and off-normal conditions and must be evaluated for protection against externally generated missiles, where the (NHS) is used during normal operation. The UHS consists of water in the isolation condenser (IC) pools, equipment pool, reactor cavity pool, and fuel pool, which support UHS function and are designed as an integral part of the RB structure. In the event of a DBA during operational modes, heat is transferred from the fuel in the reactor and containment atmosphere to the water in the IC Pools, Equipment Pool, and Reactor Cavity Pool located in the RB. The reinforced concrete walls and floors of the Reactor Building protect the pools from externally generated missiles. Based on the design and location of the pools, the staff finds that the design meets RG 1.27 as it relates to the capability of the ultimate heat sink to withstand the effects of externally generated missiles.

The new fuel and spent fuel are stored in the racks in the fuel pool, which are evaluated in accordance with RG 1.13 and defined in PSAR Section 9.1. The fuel pool is located within the Reactor Building with reinforced concrete walls and floors for protection from externally generated missiles. Therefore, the staff concludes that the fuel pool meets the guidelines of RG 1.13.

RG 1.115 describes methods acceptable for identification and protection of important safety-related SSCs from the effects of missiles generated by turbine failure. Section 3.5.1.3 of this report addresses the staff's evaluation of turbine-generated missiles and protective features to ensure SSCs are protected from equipment failure.

Based on the applicant's description of SSCs to be protected from externally generated missiles, as discussed above, the staff finds the design approach is adequate to meets the requirements of GDC 2 and 4. Additional verification of adequate protection for specific SSCs requiring protection will be verified during OL phase and review of FSAR.

3.5.2.4 Conclusion

Based on its review and the evaluation discussed above, the NRC staff finds that the information in PSAR Section 3.5.2 is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(1)(ii) and GDC 2 and GDC 4 requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.5.3 Barrier Design Procedures

3.5.3.1 Introduction

PSAR Section 3.5.3 describes the procedures by which seismic Category 1 and seismic Category RW-IIa structures and barriers are designed to resist the design-basis missiles described in PSAR Section 3.5.1 to meet the relevant requirements of GDC 2 and GDC 4. The PSAR describes procedures for local damage prediction of structures and barriers of reinforced concrete, steel, and steel-plate composite construction. The PSAR also describes energy balance methods and the nonlinear time-history dynamic analysis method used for overall global response and damage prediction of these structures and barriers. The PSAR also includes dynamic increase factors (DIF) and permissible ductility ratios for the above materials of construction. Relevant consensus standards ACI 349-13 (Appendix F), ANSI/AISC N690-18, topical report BC-TOP-9A, and other applicable references from literature used are identified along with relevant NRC guidance. The applicant amended this PSAR section by CPA Supplement 2 dated October 30, 2025 (Package: NRC 2026-TN13026; PSAR update: TVA 2025-TN13027) and CPA Supplement 5 dated January 7, 2026 (package: NRC 2026-TN13028;

PSAR update: TVA 2026-TN13029). The applicant further made editorial amendment to PSAR Table 3.5-5a by CPA Supplement 6 dated February 4, 2026 (package: NRC 2026-TN13030; PSAR update: TVA 2026-TN13031).

3.5.3.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case are for protection against dynamic effects of missiles etc. from internal and external events (including natural phenomena).
- GDC 2 requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions and consideration of appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena.
- GDC 4 requires that SSCs important to safety shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

SRP Section 3.5.3, Revision 3, "Barrier Design Procedures," issued March 2007, provides relevant guidance, including listing acceptance criteria adequate to meet the above requirements and review interfaces with SRP Section 3.5.1 for missile parameters (including turbine missiles) and SRP Section 3.5.2 for review of SSCs to be protected from external missiles. Based on PSAR Table 1.9-20, the following NRC regulatory guides (RGs) also contain relevant guidance for this review: RG 1.142, Revision 3 (NRC 2020-TN13101), and RG 1.243, Revision 0 (NRC 2021-TN13112), and RG 1.217, Revision 0 (NRC 2011-TN13103. Section 5.8.2 of NRC-approved LTR NEDC-33926P-A, Revision 4 (ML25351A085 (package); ML25351A087 (Proprietary); GE Hitachi 2025-TN13024 (Public), provides a methodology and acceptance criteria for missile impact design for local failure of steel-plate composite structures with diaphragm plates (DP-SC) structures.

3.5.3.3 Technical Evaluation

The staff reviewed PSAR Section 3.5.3 against the regulatory requirements identified above and the relevant acceptance criteria in SRP Section 3.5.3, Revision 3, and LTR NEDC-33926P-A where referenced.

Section 3.5.1, "Missile Selection and Description," of this report documents the staff's evaluation of the design missile spectrum for design of structures and barriers.

3.5.3.3.1 Local Damage Prediction

For missile impact on concrete structures and barriers, the staff noted that the applicant's procedures use empirical equations (such as modified National Defense Research Committee [NDRC] formula) provided in Kennedy, R.P., "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," Nuclear Engineering and Design, Volume 37, Issue 2, May 1976, to determine required barrier thickness to prevent perforation and scabbing, and satisfy the criteria for local impact effects in Section F.7 of Appendix F of ACI 349-13. The PSAR also specifies that the resulting thickness cannot be less than the minimum thickness necessary to protect against tornado missiles provided in PSAR Table 3.5-2, which is the same as Table 1 of SRP Section 3.5.3 for tornado Region I. The staff finds the procedures for concrete local damage acceptable because they are consistent with the acceptance criteria in Section II.1.A of SRP Section 3.5.3 and the criteria in ACI 349-13, Appendix F, as endorsed by RG 1.142.

For missile impact on steel structures and barriers, the staff noted that the minimum plate thickness to prevent perforation is obtained by increasing the more conservative thickness of the formula in C.R. Russell, "Reactor Safeguards," Macmillan, New York 1962, and the Stanford Equation in ORNL-NSIC-5, "US Reactor Containment Technology," Volume 1, Chapter 6, Oak Ridge National Laboratory, 1965, by a factor of 1.25 recommended in Topical Report BC-TOP-9-A, Revision 2, "Design of Structures for Missile Impact," September 1974. The staff finds the approach acceptable because it is based on an NRC-approved topical report and adequate to meet the acceptance criteria in Section II.1.B of SRP Section 3.5.3.

For missile impact on steel-plate composite with diaphragm plates (DP-SC) barriers, the staff noted that in PSAR Section 3.5.3.1.3, as amended by CPA Supplement 2 dated October 30, 2025 (NRC 2026-TN13026 (package), TVA 2025-TN13027 (PSAR update), the barrier is designed to prevent perforation in accordance with Section 5.8.2 of NRC-approved licensing topical report (LTR) NEDC-33926P-A, Revision 4 (ML25351A085 (package); ML25351A087 [Proprietary]); and GE Hitachi 2025-TN13024 (Public), and noted that scabbing limit state is inherently prevented by the rear steel faceplates of the steel-plate composite (SC) structure. The staff finds this approach acceptable because the methods in Section 5.8.2, "Missile Impact Design for Local Failure," of NEDC-33926P-A, Revision 4, which includes alternative empirical rational method (with specified missile parameter range of applicability) as well as explicit dynamic inelastic analysis, were reviewed and found acceptable by the NRC staff in its safety evaluation of the LTR (ML25268A140 [Proprietary]), NRC 2025-TN13032 (Public) and are subject to related limitation and condition (L&C) 8.7. Based on its evaluation in Section 3.8.1.3.2 of this report, the staff also finds that L&C 8.7 is adequately addressed and incorporated in PSAR Section 3.5.3 and Tables 3.5-5a and 3.5-5b, as amended by CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029).

The PSAR also notes that the explicit dynamic inelastic finite element analysis method may also be used in lieu of the empirical rational methods also for concrete and steel barriers. The staff also finds this approach acceptable because it is also a recommended and proven approach in industry guidance such as NEI 07-13. If used, the staff will review its implementation on a case-by-case basis during the review of the FSAR.

Based on the above review, the staff finds that the applicant's barrier design procedures for local damage prediction under missile impact meet the guidelines and acceptance criteria in Subsection II.1 of SRP Section 3.5.3 and Section 5.8.2 of LTR NEDC-33926P-A, and therefore are acceptable for preliminary design to meet relevant requirements of GDC 2 and GDC 4.

3.5.3.3.2 Overall Damage Prediction

The staff noted that the PSAR, as amended, included the following energy balance methods to determine the overall (global) response of a structure subjected to impact load where the calculated value of required ductility ratio determined by the method (maximum target displacement from missile impact divided by the component yield displacement) is shown to not exceed the applicable value of permissible ductility ratio in PSAR Table 3.5-4 for steel and reinforced concrete targets, and Tables 3.5-5a and 3.5-5b for steel-plate composite (DP-SC) targets.

a. A procedure published by Williamson, R.A., and Alvy, R.R., “Impact Effect of Fragments Striking Structural Elements,” Holmes and Narver Inc., 1973, where the impact is assumed to be plastic and deformability of the missile is not considered. The staff finds this procedure acceptable because it is included as an acceptable procedure in SRP Section 3.5.3.
b. The missile-target interaction method in Topical Report BC-TOP-9A, Revision 2, “Design of Structures for Missile Impact,” Bechtel Power Corporation, September 1974, where the deformability of the missile is accounted for in the interface forcing function and the overall structural response of the target is determined by equating the available target strain energy to the required strain energy to stop a missile. The staff finds this approach acceptable because it is based on an NRC-approved topical report, BC-TOP-9A.

The staff finds the values of permissible ductility ratios for steel and reinforced concrete in PSAR Table 3.5-4 acceptable because they are consistent with ANSI/AISC N690-18 (as endorsed in RG 1.243) and ACI 349-13 (as endorsed in RG 1.142), respectively. The staff finds the structural acceptance criteria for steel-plate composite targets in terms of allowable support rotations, ductility and allowable strains in PSAR Tables 3.5-5a and 3.5-5b, as amended by CPA Supplements 5 (TVA 2026-TN13029) and 6 (TVA 2026-TN13031), acceptable because they are consistent with LTR NEDC-33926P-A, Revision 4, as approved in NRC Safety Evaluation (ML25268A140 [Proprietary]), NRC 2025-TN13032 (Public) including related L&C 8.7.

The staff finds the dynamic increase factors (DIFs) in PSAR Table 3.5-3, based on material strain rate effects that may be applied to static material strengths of steel, concrete, and steel-plate composite constituents subject to dynamic impact loading, acceptable because they are based on consensus standards or industry documents such as ANSI/AISC N690-18, ACI 349-13 (Appendix F), NEI 07-13, endorsed in RG 1.243, RG 1.142, RG 1.217, ASCE 59-22, and NRC-approved LTR NEDC-33926P-A.

The staff also noted that the PSAR also included a second nonlinear time-history dynamic analysis method for overall response of a structure subjected to missile impact loading, where the mass and inertial properties, and nonlinear stiffness of the structural members are included in a time-history dynamic analysis. The staff finds this an acceptable method that is also recommended in industry standard NEI 07-13, endorsed in RG 1.217. If used, the staff will review the implementation of the method on a case-by-case basis as part of the review of the FSAR.

Based on the above, the staff finds that the applicant’s barrier design procedures for overall damage prediction and response under missile impact meet the guidelines and acceptance criteria in Subsection II.2 of SRP 3.5.3, and therefore acceptable for preliminary design to meet relevant requirements of GDC 2 and GDC 4.

3.5.4 Conclusion

The staff has reviewed the information provided in PSAR Section 3.5.3, amended consistent with Supplements 2, 5 and 6 (see audit summary report at NRC 2026-TN13251), and for the reasons given above, concludes that the procedures used for barrier design to withstand design-basis missile impact are acceptable to meet the relevant requirements of GDC 2 and 4 with respect to the procedures used to demonstrate capabilities of the structures, shields, and barriers to provide sufficient protection to SSCs that must withstand the effects of environmental conditions including natural phenomena (tornado and hurricane missiles) and environmental dynamic effects including the effects of missiles, pipe whipping, and discharging fluids. Therefore, the NRC staff finds that the information provided by the applicant is sufficient for preliminary design is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(1)(ii) and other regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35. The staff will confirm the implementation of the barrier design procedures during the evaluation of the CRN-1 FSAR.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.6.1.1 Introduction

This section evaluates the design bases and criteria relied upon to demonstrate that essential SSCs are protected against postulated piping failures. It identifies high- and moderate-energy systems representing potential sources of dynamic and environmental effects associated with pipe rupture and defines the criteria for the separation and evaluation of adverse consequences.

The applicant indicated that the final analysis of pipe break events will be performed in the Final Safety Analysis Report (FSAR) to identify essential SSCs to be protected from postulated piping failures. The staff will evaluate the final analysis of pipe break events during the review of the operating license.

Section 3.6, "Protection against Dynamic Effects Associated with the Postulated Rupture of Piping," of the PSAR discusses information related to protection against pipe rupture effects. Table 3A-1, "Preliminary BWRX-300 Component Classification List," identifies BWRX-300 SSCs, their location, seismic category, and Quality Group.

3.6.1.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is related to demonstrating that essential SSCs are protected against postulated piping failures.
- GDC 2, as it requires the protection of SSCs important to safety to withstand the effects of natural phenomena, such as earthquakes.
- GDC 4, as it requires SSCs important to safety to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with postulated pipe rupture.

The guidance in SRP Section 3.6.1, Revision 3, issued March 2007, provides relevant regulatory guidance, as well as review interfaces with other SRP sections.

3.6.1.3 Technical Evaluation

In PSAR Section 3.6.1, the applicant described the methodology used in designing the protections for the essential systems and components from the consequences of postulated piping failures. The applicant defined essential systems and components as those SSCs that are required to shut down the reactor and to mitigate the consequences of the postulated piping rupture.

In PSAR Section 3.6.1.1.5, “Protection of Essential Structures, Systems, and Components,” the applicant indicates that if the consequence of dynamic effects on essential SSCs is unacceptable, the following mitigation options may be applied:

1. The essential SSCs are relocated.
2. The whipping pipe is restrained by a whip restraint, resulting in a smaller restrained zone of influence that does not encompass the essential target.
3. A protective barrier or jet shield is used to protect the essential SSCs from impact (barrier) or jet impingement (barrier or jet shield).
4. The essential SSCs are qualified to operate following the pipe rupture interaction.

In PSAR Section 3.6.1.2, “Definition of High Energy and Moderate Energy Fluid Systems,” the applicant defines high-energy fluid systems as those systems or portions of systems that, during normal plant conditions, are either in operation or are maintained pressurized under conditions where either or both of the following are met:

- Maximum operating temperature exceeds 200°F.
- Maximum operating pressure exceeds 275 psig.

Systems or portions of systems pressurized above atmospheric pressure during normal plant conditions that are not classified as high energy are classified as moderate-energy fluid systems.

The reviews of previous nuclear power plant designs indicated that the functional or structural integrity of systems and components required for safe shutdown of the reactor and maintenance of cold shutdown conditions could be endangered by fluid system piping failures at locations outside containment. The staff has identified an acceptable approach for the design and arrangement of fluid systems located outside of containment to ensure that the plant can be safely shut down in the event of piping failures outside containment. SRP Branch Technical Position (BTP) 3-3, Revision 3, “Protection against Postulated Piping Failures in Fluid Systems Outside Containment,” issued March 2007, and its companion BTP 3-4, Revision 2, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” issued March 2007, describe this approach.

In PSAR Table 1.9-3, “Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems),” the applicant indicated that they are in conformance with the guidance indicated in SRP Section 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment”; SRP Section 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping”; BTP 3-3, “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment”; and BTP 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.”

The staff evaluated the applicant's definitions of high- and moderate-energy systems and found them to be consistent with the definitions provided in BTP 3-3, which delineates the staff's guidelines for protection against postulated piping ruptures in fluid systems outside the containment. The staff finds the system definitions above to be acceptable.

General Design Criterion 2

The requirements in 10 CFR Part 50, Appendix A, GDC 2, state that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes. During a seismic event, it is postulated that non-seismic SSCs could fail. This section evaluates the impact of full circumferential ruptures of non-seismic moderate-energy piping in areas close to SSCs important to safety where the effects of a failure are not already bound by failures of high energy piping. Acceptance criteria are based on conformance to SRP BTP 3-3.

In PSAR Section 3.6.1.3, "Design (Safety) Evaluation," the applicant states that "essential SSCs are protected against postulated piping failures. The effects of postulated breaks and cracks include pipe whipping, jet impingement, fluid decompression waves within the ruptured pipes and environmental effects such as temperature, pressure, humidity, chemical exposure, radiation, spray wetting, flooding, and sub-compartment pressurization." In Rev. 1 of PSAR Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," the applicant states that "break and crack location criteria and methods of analysis for dynamic effects are discussed in this subsection in accordance with NUREG-0800, SRP 3.6.2." Table 1.9-3 states that for the topic of failures of fluid system piping outside containment affecting safe shutdown equipment, the plant is designed to be in conformance with BTP 3-3.

PSAR Section 3.6.1.2 includes a list of potential candidates for a postulated pipe break during normal plant conditions that are analyzed for potential damage resulting from damage effects and potential candidates for a postulated pipe crack. The final pipe break event evaluation that will be performed in the FSAR will verify that the consequences of failures of high- and moderate-energy lines do not affect the ability to safely shut down the plant.

The staff finds that the applicant identified that the equipment that requires protection from failures of fluid system piping will be evaluated in accordance with the recommendations of BTP 3-3, which the staff identified as the acceptable guidance to demonstrate conformance with GDC 2. Therefore, the staff finds that the above system description is acceptable and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a(3)(ii).

General Design Criterion 4

The plant design for protection against postulated piping failure in fluid systems must meet the requirements of GDC 4 as it relates to accommodating the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids. These requirements are imposed to ensure that (1) piping failures in fluid systems will not cause the loss of needed functions in safety-related systems, and (2) the plant could be safely shut down in the event of such a failure.

In PSAR Section 3.1.1.4, GDC 4, "Environmental and Dynamic Effects Design Bases," the applicant indicates that all Safety Class (SC) 1 SSCs are designed to be compatible with environmental conditions associated with postulated pipe failure accidents. The SC2 and SC3 SSCs that mitigate potentially induced events are confirmed to be compatible with the environmental and dynamic effects present during the event scenarios where these functions are credited. Additionally, the PSAR indicates that failure of non-SC1 equipment due to environmental and dynamic effects will not impair SC1 equipment.

For the piping in containment penetration areas, the applicant classified these areas as break exclusion zone in accordance with BTP 3-4 B.1(ii). Sections of piping systems in containment penetration areas that conform with BTP 3-4, are exempted from the evaluation of the dynamic effects of postulated breaks in high-energy piping, and the installation of the corresponding whip restraints and jet shields.

In PSAR Section 3.6.1.1.1, the applicant defined the boundaries of the break exclusion zone starting from the piping weld at the flanged connection to reactor isolation valves (RIVs) to the outboard containment isolation valves (CIVs) and past the outboard CIVs to the nearest seismic interface restraint. The RIVs act as the inboard CIV for all high-energy systems subjected to BTP 3-4 B.1(ii) criteria. In Licensing Topical Report (LTR) NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” the applicant also states that a terminal end pipe break would be postulated at the piping connection to the reactor pressure vessel outboard isolation valve as specified in BTP 3-4, B.1(iii)(2).

Also, in LTR NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” the applicant states that the **[[**

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]]. The NRC Staff evaluated LTR NEDC 33910P-A and issued “USNRC Safety Evaluation Report for NEDC-33910P-A” (GE Hitachi 2021-TN13033). Section 4.1.6 of this report states, in part, “The NRC staff finds this approach, as described in NEDC-33910, consistent with 10 CFR Part 50, Appendix A, GDC 4, and, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 4 is satisfied when an application for a BWRX-300 SMR is received.” In PSAR Section 5.4.3, the break exclusion area requirements for the area between the piping weld between at the flanged connection to the RIVs and the RPV are further discussed.

BTP 3-4 is an acceptable methodology for demonstrating conformance with GDC 4, and applying these criteria to limit the locations where postulated breaks can occur is acceptable. In Section 3.6.2 of this report, the staff evaluates the applicant’s implementation of the pipe break location methodology.

In LTR NEDC-33911P-A, “BWRX-300 Containment Performance,” the applicant provided the design requirements, analytical methods, acceptance criteria, and regulatory bases for the containment performance design functions of the BWRX-300 small modular reactor. In PSAR Section 3.6.1.1.1, “Break Exclusion Zone,” the applicant references LTR NEDC-33911P-A, which addresses the break exclusion zone boundaries. The LTR and the PSAR describe the Isolation Condenser System (ICS), indicating that the ICS does not have an outboard containment isolation valve and all of the ICS piping is within the break exclusion zone.

The NRC staff evaluated LTR NEDC-33911P-A and issued "USNRC Safety Evaluation Report for NEDC-33911P-A" (GE Hitachi 2021-TN13034 – public version). In Section 6 of the staff's report, the staff identified two limitations and conditions. Condition 2 states that:

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The applicant has not completed these evaluations at this time. The evaluations will be reviewed at the time of FSAR review.

In PSAR Section 3.6.1.1.3, "Augmented Inservice Inspections," the applicant indicates they will pre-examine piping welds and base metal in the break exclusion zone and perform ISI in accordance with ASME Boiler and Pressure Vessel Code (BPVC, Section XI, Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants" (Reference 3.6-3). The applicant will also perform ISI on 100 percent of the welds during each inspection interval. The staff finds this acceptable because it meets NUREG-0800 (NRC 2021-TN8013), BTP 3-4 B.1(ii)(7).

In PSAR Section 3.6.1.1.4, "Guard Pipe Assembly Design Criteria," the applicant indicates where guard pipes are used. The enclosed portion of fluid system piping will utilize seamless construction without circumferential welds unless specific access provisions are made to permit in-service volumetric examination of the longitudinal and circumferential welds. The staff finds this acceptable because it meets NUREG-0800 (NRC 2021-TN8013), BTP 3-4, B.1.(ii)(3).

In addition, PSAR Section 3.6.1.1.4 describes that the guard pipe portions provided for the piping in the containment penetration area are designed in accordance with the criteria of the ASME BPVC, Section III, Subsection NE, "Class MC Components," where the guard pipe is part of the containment boundary. Also, the guard pipe assembly is designed to meet the following design criteria and tests:

1. The design pressure and temperature are not less than the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions.
2. The Level C stress limits in NE-3220 are not exceeded under the loading associated with containment design pressure and temperature in combination with the SSE.
3. Guard pipe assemblies are subjected to a single pressure test at a pressure not less than its design pressure.
4. Guard pipe assemblies do not prevent the access necessary to conduct the in-service examination specified in BTP 3-4, 2.A(ii)(7).

Inspection ports, if used, are not located in that portion of the guard pipe through the annulus of dual barrier containment structures.

The staff finds this acceptable because it meets NUREG-0800 (NRC 2021-TN8013), BTP 3-4, B.1.(ii)(6).

[1] FMCRD = fine motor control rod drive.

In BTP 3-3, Section B, Item 1.a (1), the staff indicates that, even though portions of the main steam and feedwater (FW) lines meet the break exclusion requirements of BTP 3-4, Item B.1.(ii), essential equipment must be protected from an assumed nonmechanistic longitudinal break with a cross-sectional area of at least 929 square centimeters (cm^2) (1 ft^2). This failure is postulated to establish the environmental conditions that the essential SSCs need to be protected (or designed) against.

In PSAR 3.6.1.1.2, “Environmental Effects in Break Exclusion Zone,” the applicant indicates that a nonmechanistic longitudinal break with a cross-sectional area of 1.0 ft^2 (0.0929 m^2) is disproportionately large for a small modular reactor with small pipe sizes. The applicant proposes to proportionally scale the nonmechanistic longitudinal break cross-sectional area to reflect the smaller main steam and feedwater pipe size used in the BWRX-300 design. The final proportionally scaled nonmechanistic longitudinal break will be provided in the FSAR.

The staff has previously recognized that a nonmechanistic longitudinal break with a cross-sectional area of 1.0 ft^2 (0.0929 m^2) can disproportionately be large for a small modular reactor with small pipe sizes. Therefore, the staff finds it acceptable to scale the nonmechanistic longitudinal break according to the final pipe size used in the BWRX-300 design. At the time of the FSAR review, the staff will review the proposed nonmechanistic longitudinal break cross-sectional area and the protection of the essential SSCs from the postulated nonmechanistic longitudinal break.

The staff reviewed the information in the PSAR, LTR NEDC-33910P-A, and LTR NEDC 33911P A and determined that the applicant has proposed to follow the guidance provided in NUREG-0800 (NRC 2021-TN8013) SRP Sections 3.6.1 Revision 3 and 3.6.2 Revision 3, BTP 3-3 Revision 3, and BTP 3-4 Revision 3. The staff finds that by designing piping systems in accordance with the recommendations of BTP 3-4 Revision 3, the applicant has reduced the likelihood of -highenergy- failures and thereby protected the essential SSCs from a postulated highenergy line failure. By designing the essential SSCs to the anticipated environmental conditions resulting from a nonmechanistic- pipe failure, the applicant will protect the essential SSC functions against a nonmechanistic pipe failure in the break exclusion zone in conformance with GDC 4. Therefore, staff finds that the above system description is acceptable and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a(3)(ii).

3.6.1.4 Conclusion

The staff has reviewed the available information provided in the PSAR, and, for the reasons given above, concludes that the applicant has identified an acceptable methodology for designing and evaluating the protection against postulated piping failures outside containment. With respect to accommodating the environmental effects of postulated pipe ruptures by considering the environmental effects from the rupture of non-seismic piping, the facility is designed conforming to BTP 3-3. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other regulatory requirements contained in GDC 2 and GDC 4 and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2.1 Introduction

In PSAR Section 3.6.2, the applicant provides the break and crack location criteria and methods of analysis for dynamic effects to evaluate the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping inside and outside of the primary containment in accordance with NUREG-0800 (NRC 2021-TN8013), SRP Section 3.6.2. The information in this section provides the basis for the requirements for the protection of essential SSCs.

3.6.2.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is related to the adequate protection provided such that the effects of the postulated pipe breaks do not adversely affect the functionality of SSCs relied upon for safe reactor shutdown and that the consequences of the postulated pipe rupture are mitigated.
- In 10 CFR Part 50, Appendix A, GDC 4, the NRC requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including lost-of-coolant accidents.

Specific SRP acceptance criteria and guidance documents acceptable to meet the requirements of the NRC regulations identified above are as follows for this review:

- SRP Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects," Revision 3
- BTP 3-4, Revision 3, contains the staff's guidelines for defining postulated rupture locations in fluid system piping inside and outside the containment.

3.6.2.3 Technical Evaluation

In PSAR Section 3.6.2.1.1, "Location of Postulated Break," the applicant indicated, for the piping in the containment penetration area, the dynamic effects resulting from the postulation of guillotine breaks in high-energy lines are excluded based on meeting the criteria in BTP 3-4. The staff reviewed this information and finds it is consistent with BTP 3-4, B.1.(ii)(1) through B.1.(ii)(7) as well as the design criteria of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Subarticle NE-1120 as stated in BTP 3-4.

For the ASME BPVC Section III class 1 piping in areas other than the containment penetration, the applicant indicated the pipe breaks are postulated at terminal ends and at intermediate locations where the maximum stress range or fatigue usage values exceed the limits specified in BTP 3-4. The staff reviewed this information and finds it is consistent with BTP 3-4, B.1.(iii)(1)(a) through B.1.(iii)(1)(c).

For the ASME BPVC Section III class 2 and 3 piping in areas other than containment penetration, the applicant indicated the pipe breaks are postulated at terminal ends and at intermediate locations where the maximum stress values exceed the limits specified in BTP 3-4. The staff reviewed this information and finds it is consistent with BTP 3-4, B.1.(iii)(2)(a) and B.1.(iii)(2)(b).

For the non-ASME BPVC Section III piping, the pipe breaks in seismically analyzed high-energy non-ASME Section III piping are postulated according to the same criteria as for ASME BPVC Section III, Class 2 and 3 high-energy piping systems. The pipe breaks in non-seismically analyzed high-energy non-ASME BPVC Section III piping are postulated at terminal ends and at each intermediate location such as pipe fittings, valves, flanges, and welded attachments. The staff reviewed this information and finds it is consistent with BTP 3-4, B.1.(iii)(3).

In PSAR 3.6.2.1.2, "Locations of Postulated Pipe Cracks," the applicant indicated that, for the high- and moderate-energy piping associated with the containment penetration area, they considered analytical concepts to eliminate the need to consider postulated cracks as described in BTP 3-4. The staff reviewed this information and, for the high-energy piping, finds it is consistent with BTP 3-4, B.1.(v) and for the moderate-energy piping is consistent with BTP 3-4, B.2.(ii).

For the ASME BPVC Section III Class 1, 2, 3 and non-ASME BPVC Section III high-energy piping in areas other than containment penetration areas, the applicant indicated that the pipe cracks are postulated as follows:

- For ASME Class 1 piping at axial locations where the calculated stress range values exceed the limits specified in BTP 3-4.
- For ASME Class 2 and 3 or non-ASME BPVC Section III piping at axial locations where the calculated stress values exceed the limits specified in BTP 3-4.
- For non-seismically analyzed non-ASME BPVC Section III piping that has not been evaluated to obtain stress information, through-wall cracks are postulated at axial locations that produce the most severe environmental effects.

The staff reviewed this information and, for the high-energy piping, finds it is consistent with BTP 3-4, B.1.(v)(1) through B.1.(v)(3).

For the ASME BPVC Section III Class 1, 2, 3 and non-ASME BPVC Section III moderate-energy piping in areas other than containment penetration areas, the applicant indicated that the pipe cracks are postulated where

- For ASME BPVC Class 1 piping, the calculated stress range values exceed the limits specified in BTP 3-4.
- For ASME BPVC Class 2 or 3 and non-ASME BPVC Section III piping, the calculated stress values exceed the limits specified in BTP 3-4.

In addition, through-wall cracks, unless the piping system is excluded above, are postulated at axial and circumferential locations that result in the most severe environmental effects. The staff reviewed this information for the moderate-energy piping and finds it consistent with BTP 3-4, B.2.(iii)(1) through BTP 3-4, B.2.(iii)(3).

PSAR Section 3.6.2.1.3, "Types of Breaks and Cracks to be Postulated," lists the criteria for postulating high-energy circumferential and longitudinal pipe breaks at specific locations. The staff reviewed this information for the circumferential and longitudinal pipe breaks and finds it consistent with BTP 3-4, B.3.(i)(1) through BTP 3-4, B.3.(i)(5) and BTP 3-4, B.3.(ii)(1) through BTP 3-4, B.3.(ii)(5).

PSAR Section 3.6.2.1.3 also describes the criteria used to postulate through-wall leakage cracks in high- or moderate-energy fluid system piping at specific locations. The staff reviewed this information for the pipe cracks and find it consistent with BTP 3-4, B.3.(iii)(1) through BTP 3-4, B.3.(iii)(4).

In PSAR Section 3.6.2.2.1, "Analytical Methods to Define Blowdown Forcing Functions," the applicant describes the thrust force and its magnitude are developed by analytical solutions (closed-form equations) or by numerical simulation (computational fluid dynamics techniques) for the postulated circumferential and longitudinal breaks. The applicant will use analytical methods used to establish pipe rupture blowdown and jet thrust forcing functions in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.2, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," Appendix B (Reference 3.6-7), modified by NUREG/CR-7275, *Jet Impingement in High-Energy Piping Systems*, to address SRP Section 3.6.2, Appendix A, "Potential Nonconservatism of ANSI/ANS 58.2 Standard's Jet Modeling." The staff did not verify application of the methodology as these details were not provided in the PSAR. However, the staff understands as indicated in PSAR Section 3.6.2.2.1 that methodology will meet SRP Section 3.6.2 including Appendix A. The staff will review the implementation of the methodology on a case-by-case basis as part of the review of the FSAR as described in SRP Section 3.6.2, Appendix A.

PSAR Section 3.6.2.2.2, "High Energy Pipe Whip Analysis," describes the applicant's considerations for pipe whip analysis, which are the following:

- Two ends of the broken pipe move clear of each other unless physically limited by piping restraints, structural members, or piping stiffness.
- Determine the direction of the thrust force, axial to the broken pipe segment (perpendicular to the break area).
- The unrestrained zone of influence is determined in the direction of the jet reaction initially, with the total path controlled by the piping geometry.
- If unrestrained, a whipping pipe having a constant energy source sufficient to form a plastic hinge is considered to form a plastic hinge and rotate about the plastic hinge or the nearest rigid pipe whip restraint, anchor, or wall penetration capable of resisting the pipe whip loads; or rotate about the calculated dynamic plastic hinge location.
- Where the hinge moment has a torsional and a bending component, the hinge does not result in a planar unrestrained zone of influence. In this case, a half-sphere unrestrained zone of influence is assumed with the pipe sweeping a 180° arc from its initial position to its final resting position, unless a more detailed analysis is performed to justify a smaller unrestrained zone of influence.
- If the plane of motion of a whipping pipe is normal to a flat surface at impact, it is assumed that the pipe comes to rest against that surface.

- The internal fluid energy level associated with the pipe break reaction considers any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe is considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.
- Credit is taken for the limited amount of energy in closed-ended piping runs near dead ends or normally closed valves.
- For those portions of high-energy piping systems that are normally pressurized only during plant conditions other than 100 percent power, the thermodynamic state that produces the most severe fluid reaction forces is used.

Based on the above approaches the staff finds that the pipe whip analysis methods and considerations described in the PSAR acceptable because the analytical methods are appropriate and consistent with the methods recommended in SRP Section 3.6.2.

In PSAR Section 3.6.2.3.1, "Jet Impingement Analyses," the applicant describes that the jet geometry modeling and the calculation of the jet impingement force acting on a target will be in accordance with ANSI/ANS 58.2, with modifications applied as identified in NUREG/CR-7275 to address nonconservatisms identified in SRP Section 3.6.2, Appendix A. The staff did not verify application of the methodology as these details were not provided in the PSAR. However, the staff understands as indicated in PSAR Section 3.6.2.2.1 that methodology will meet SRP Section 3.6.2 including Appendix A. The staff will review the implementation of the methodology on a case-by-case basis as part of the review of the FSAR as described in SRP 3.6.2, Appendix A.

PSAR Section 3.6.2.3.2, "Design Codes and Load Combinations for Pipe Whip Restraint," describes the applicant's approach to pipe whip restraint design, which is the following:

- Pipe whip restraints that only act to restrain the effects of pipe breaks, and are not active for other loads, are designed to ANSI/American Institute of Steel Construction (AISC) N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities" (Reference 3.6-8). They are not designed to ASME BPVC, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," Subsection NF, "Supports."
- Pipe whip restraints that also act as restraints for other loads (e.g., deadweight, seismic) are designed to ASME BPVC, Section III, Subsection NF (Reference 3.6-9). However, the qualification for the pipe break impact load follows ANSI/AISC N690-18.
- The pipe whip restraints are designed to allow access for ISIs. The measures taken for the protection of essential SSCs do not preclude the conduct of in-service examinations of pressure-retaining components as required by the rules of ASME BPVC Section XI.
- The pipe-restraint gap is sufficiently small to limit the kinetic energy of the whipping pipe and therefore the impact load.
- The pipe-restraint gap is sufficiently large to prevent interference from thermal expansion of the line.
- Pipe whip restraints are located to prevent the formation of a plastic hinge.

The pipe whip restraints, jet shields, and barriers are designed for the following loading conditions and load combinations (Service Level D):

$$DW + TH + \sqrt{DBPB^2 + (|SSE| + |SSES| + |SWE|)^2}$$

Where:

DBPB = Design Basis Pipe Break (including jet impingement)
DW = Deadweight
SSE = Safe Shutdown Earthquake
SSES = SSE seismic anchor motion of the restraint structure
SWE = Seismic Self-Weight excitation of the restraint structure
TH = thermal loads due to the expansion of the whip restraint relative to the post-break hot building structure.

The applicant classifies the pipe whip restraints as seismic Category II, and they are seismically designed to prevent adverse interaction with seismic Category I SSCs.

Based on the description above , the staff finds that the pipe whip restraint methods described in the PSAR are acceptable because the considerations are appropriate and the methods are consistent with the methods recommended in SRP Section 3.6.2.

PSAR Section 3.6.2.4, “Analytical Methods to Define Blast Wave Interaction to Essential SSCs.

Blast Wave Effects, provides the following considerations:

- High-energy blast wave analysis is performed assuming the break opening time as instantaneous, maximizing blast formation.
- The formation and effects of a blast wave caused by a high-energy line break (HELB) is evaluated using three-dimensional computational fluid dynamics modeling technique that reflects the thermo-hydraulic parameters at the instant of the postulated pipe rupture, within 1 millisecond of the onset of the break.
- These blast wave forces on target surfaces are impulsive loads that last a few milliseconds or less.
- The blast analysis includes amplification caused by blast wave reflections against walls, and the effects of angled incidence.
- Because of pressure-relieving clearing at the edges of a target surface, and the short duration of the pressure impulse, small structures are not exposed to significant loading. The analysis of blast wave effect is therefore limited to walls, floors, and large vessels.
- Blast force and impulse may be bounded by the jet thrust forces that subsequently develop.

The staff did not verify application of the methodology as these details were not provided in the PSAR. However, the staff understands that the methodology will meet SRP Section 3.6.2, Appendix A. The staff will review the implementation of the methodology on a case-by-case basis as part of the review of the FSAR as described in SRP Section 3.6.2, Appendix A.

In Section 3.6.2.5, the applicant indicated that a summary of the dynamic analyses applicable to high-energy piping systems will follow the guidelines provided in SRP Section 3.6.2. The staff will review this information during the review of the operating license.

As discussed above, PSAR Section 3.6.2 meets the guidance in SRP 3.6.2 and BTP 3-4, and, therefore, also meets the relevant requirements in 10 CFR 50.34(a)(3)(ii) and 10 CFR Part 50, Appendix A, GDC 4.

3.6.2.4 Conclusion

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the applicant's description of the plan to postulate pipe ruptures and to design SSCs to accommodate and protect against the associated dynamic effects has met the relevant requirements of GDC 4. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.7 Seismic Design

3.7.1 Seismic Design Parameters

3.7.1.1 Introduction

PSAR Section 3.7, "Seismic Design," states that the containment structure, the containment internal structures, and the reactor building (RB) enveloping them are the only seismic Category 1 civil structures for CRN-1. The radwaste building (RWB) is seismic Category RW-IIa per RG 1.143. The power block structures, namely, the control building (CB), turbine building (TB), service building, and reactor auxiliary structures have potential to interact with the RB during seismic or extreme wind events and are therefore seismic Category II.

PSAR Section 3.7.1, "Seismic Design Parameters," describes information on the following design parameters used as input to the seismic analysis and design of seismic Category 1 structures for CRN-1:

- design ground motion (design response spectra and design time histories)
- percentage of critical damping values
- supporting media for seismic Category 1 structures

Relevant consensus standards ASCE 4-16, "Seismic Analysis of Safety-Related Nuclear Structures," ASCE/SEI 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities"; licensing topical report NEDO-33914-A, "BWRX-300 Advanced Civil Construction and Design Approach"; and other applicable references from literature used are identified along with relevant NRC guidance.

3.7.1.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), requires applicants to address the design bases (in this case, the development of seismic design parameters) and the relation of the design bases to the principal design criteria, which in this case, these seismic parameters are used to meet GDC 2 for protection against earthquake natural phenomena.

- GDC 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions, with consideration of (1) the most severe natural phenomena historically reported for the site and surrounding area, (2) appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena, and (3) the importance of the safety functions to be performed.
- 10 CFR Part 50, Appendix S, requires in part that, for safe shutdown earthquake (SSE) ground motions, SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after vibratory ground motion through design, testing, or qualification methods. The evaluation must account for soil-structure interaction (SSI) effects and the expected duration of the vibratory motion. If the operating-basis earthquake (OBE) is set at one-third or less of the SSE, an explicit analysis or design is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied. Appendix S also requires the horizontal component of the SSE ground motion in the free field at the foundation level of the structures to be an appropriate response spectrum with a peak ground acceleration of at least 1/10 the acceleration of gravity (0.1g).

SRP Section 3.7.1, Revision 4, “Seismic Design Parameters” issued December 2014, lists acceptance criteria for seismic design parameters adequate to meet the relevant requirements above, and provides review interfaces with SRP Sections 2.5.1 through 2.5.3 for review of geological and seismological information to establish the free-field ground motion, SRP Section 2.5.4 for geotechnical parameters and methods employed in the analysis of free-field soil media and soil properties, and SRP Sections 3.7.2 and 3.7.3 for seismic system and subsystem analysis.

Based on PSAR Table 1.9-20, the following NRC regulatory guides (RGs) and other guidance documents also contain relevant guidance for this review.

- RG 1.60, Revision 2, *Design Response Spectra for Seismic Design of Nuclear Power Plants*, issued July 2014, for determining the acceptability of design response spectra for input into the seismic analysis of nuclear power plants (NRC 2014-TN5937)
- RG 1.61, Revision 2, *Damping Values for Seismic Design of Nuclear Power Plants*, issued December 2023, for determining acceptability of damping values used in the dynamic seismic analyses of seismic Category 1 SSCs (NRC 2023-TN13035)
- RG 1.208, Revision 0, *A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion*, issued March 2007 for the development of the site-specific SSE ground motion response spectrum as a characterization of seismic hazard (NRC 2007-TN5858)
- DC/COL-ISG-017, “Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil-Structure Interaction Analyses,” issued March 24, 2010 (NRC 2010-TN13036)
- NUREG/CR-5347, *Recommendations for Resolution of Public Comments on USI A-40, “Seismic Design Criteria”*, issued June 1989, for determining the acceptability of the development of target power spectral density functions (NRC 1989-TN13037)
- NUREG/CR-6728, *Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines*, issued October

2001, for determining the acceptability of the ground motion characteristics (McGuire et al. 2001-TN5861)

- NEDO-33914-A, Revision 2, "BWRX-300 Advanced Civil Construction and Design Approach," Licensing Topical Report (GE Hitachi 2022-TN13022)

3.7.1.3 Technical Evaluation

The staff reviewed PSAR Section 3.7.1, regarding seismic design parameters used as input to the seismic analysis to develop seismic demands for design, against the regulatory requirements identified above and the relevant acceptance criteria in SRP Section 3.7.1, Revision 4. The staff reviewed information on (1) the design ground motion (response spectra and time histories), (2) percentage of critical damping values, and (3) supporting media for seismic Category I structures, against the acceptance criteria in SRP Section 3.7.1 to assure that these seismic design parameters used in the CRN-1 seismic analysis are adequately defined to form a conservative basis for the preliminary design of seismic Category 1 SSCs to withstand seismic loadings consistent with GDC 2.

The staff evaluation of related PSAR Sections 2.5.1 through 2.5.4 are, respectively, documented in Sections 2.5.1 through 2.5.4 of this report. This section of this report describes the results of the staff's technical evaluation of PSAR Section 3.7.1. Section 3.7.2 of this report presents the staff's evaluation of the seismic system analysis of the CRN-1 seismic Category I structures. Section 3.7.3 of this report presents the staff's evaluation of the seismic subsystem analysis for the CRN-1 substructures and subsystems.

3.7.1.3.1 Design Ground Motion

The PSAR states that the design considers the operating-basis earthquake (OBE) and safe shutdown earthquake (SSE) design ground motions, with the OBE set at one-third the SSE. Therefore, an explicit response analysis or design load combinations that consider OBE loads are not required except for the design of containment metal (Class MC) components for post-flooding and cyclic loading (fatigue) considerations. The staff finds that, with the specification of OBE as one-third of the SSE, exclusion of explicit seismic analysis and design for the OBE (with the exceptions above) is acceptable because it is consistent with 10 CFR Part 50, Appendix S, requirements, at which level the OBE is only a threshold for plant shutdown. As such, seismic system and subsystem analyses for OBE are not performed or discussed in PSAR Sections 3.7.2 and 3.7.3, and staff evaluation of seismic analysis for OBE is not provided in Sections 3.7.2 and 3.7.3 of this report.

3.7.1.3.1.1 *Design Response Spectra*

The staff reviewed the design response spectra information in PSAR Section 3.7.1.1.1 against the acceptance criteria in paragraph II.1.A of SRP Section 3.7.1. The staff noted the following horizontal and vertical response spectra, obtained from the results of the probabilistic site response analysis considering as-built subgrade conditions (engineered backfill on existing fill, in-situ residual soil, weather rock, and base rock) and evaluated in Section 2.5.2 of this report, define the CRN-1 site-specific SSE ground motion:

- Foundation input response spectra (FIRS), which define the 5 percent damped SSE ground motion of the integrated reactor building (IRB) foundation and are used as input to the SSI analyses, are shown in PSAR Figure 3.7-3 along with a comparison to RG 1.60 horizontal and vertical spectra anchored at 0.1g. PSAR Figure 3.7-3 shows the peak ground accelerations (PGAs or zero period accelerations [ZPAs]) of the horizontal and vertical design FIRS are approximately 0.38g and 0.33g, respectively.

- Performance-based surface response spectra (PBSRS), which define the 5 percent damped SSE ground motion at the finished plant grade elevation (EL 814.5 ft), are shown in in PSAR Figure 3.7-5.
- Performance-based intermediate response spectra (PBIRS), which define the 5 percent damped SSE ground motion at intermediate embedment depth elevation (at the top of un-weathered rock layer), are shown in in PSAR Figure 3.7-4. The PBIRS are used to assure that ground motion used as input to the SSI analysis of deeply embedded IRB foundation is adequate through the depth of embedment.
- The above design response spectra (FIRS, PBSRS, PBIRS) figures are extracted from PSAR Figures 2.5.2-93R, 2.5.2-94R, and 2.5.2-95R.
- The vertical FIRS, PBSRS, and PBIRS at the CRN-1 site are developed by applying vertical to horizontal (V/H) ratios calculated using publication Pezeshk, S., Farhadi, A., and Haji-Soltani, A., "A new model for vertical-to-horizontal response spectral ratios for central and eastern North America," Bulletin of the Seismological Society of America, vol 112, p 2018–2030, 2022 (PSAR Reference 3.7-4) to the horizontal spectra. The median V/H ratios are shown in PSAR Figure 2.5.2-94R.
- The frequency dependent factors shown in PSAR Figure 3.7-1 are used to augment the FIRS to ensure the vertically propagated SSI foundation input motions envelope the horizontal and vertical motions (PSBRS) and achieve hazard-consistent seismic input as recommended in the guidance in DC/COL-ISG-017 for deeply embedded foundation. The corresponding Nuclear Energy Institute (NEI) check of the augmented horizontal and vertical FIRS shown in PSAR Figure 3.7-2 indicates the envelope of the 5 percent damped spectra for the responses at the surface for the lower bound (LB), best estimate (BE), and upper bound (UB) soil profiles used for SSI analysis generally envelope the PSBRS, thereby meeting the DC/COL-ISG-017 requirement.

The staff finds that the site-specific design basis seismic response spectra defined above meet the acceptance criteria in SRP 3.7.1.II.1.A and DC/COL-ISG-017 because of the following reasons:

- The CRN-1 site-specific FIRS, PBIRS, and PSBRS are determined as free-field outcrop motions at the foundation, intermediate, and surface levels.
- The minimum PGA for the horizontal component of the SSE at the foundation level in the free field is approximately 0.38g from PSAR Figure 3.7-3, which is higher than 0.1g, thereby satisfying the minimum PGA requirement for SSE ground motion in 10 CFR Part 50, Appendix S.
- The FIRS ground motion satisfied the NEI check as shown in PSAR Figure 3.7-2 and meets the guidance in DC/COL-ISG-017 for the deeply embedded IRB foundation, thereby assuring the ground motion used as input to the SSI analysis is hazard consistent with the site response analysis.
- The use of the model in Pezeshk et al. is reasonable and acceptable because it is based on analysis of available data specific to central and eastern North America (CENA) applicable to the CRN-1 site location and comparison of predicted V/H ratios with other recently published models showed the model describes the data well and has better performance than other models developed for other study regions over the entire frequency range.

- The site-specific design response spectra were developed based on probabilistic site response analysis considering as-built subgrade conditions evaluated in Section 2.5.2 of this report.

3.7.1.3.1.2 *Design Time Histories*

The staff reviewed the design time histories information in PSAR Section 3.7.1.1.2 against the acceptance criteria in paragraph II.1.B of SRP Section 3.7.1, and noted the following:

- Acceleration time histories (ATHs) used as input to the seismic SSI analysis of the IRB are developed by spectral matching of seed ground motion records to the 5 percent damped target design ground motion FIRS shown in PSAR Figure 3.7-3.
- Five sets of three design motion ATHs (two horizontal and one vertical) are developed in accordance with the guidance in Section 5.2.3 of NRC-approved licensing topical report (LTR) NEDO-33914-A, and Option 2, “Multiple Sets of Time Histories,” in Section II.1.B of SRP Section 3.7.1.
- Details of the five sets of seed time history records, with different magnitudes and distance bins, selected from the NGA-West 2 ground motion database are provided in PSAR Table 3.7-2.
- Using the selected seed time histories in PSAR Table 3.7-2, time histories compatible to the design FIRS in PSAR Figure 3.7-3 over the frequency range of interest are generated using the time domain spectral matching method in publication by Alik, L.A., and Abrahamson, N., “An Improved Method for Nonstationary Spectral Matching,” Earthquake Spectra, vol 26, p 601–617, August 2010. The time step of the modified time histories are refined to 0.0025 for accurately calculating high-frequency responses in accordance with Section 5.2.3 of LTR NEDO-33914-A.
- Response spectra of the generated ATHs are computed at a minimum of 100 points uniformly spaced over the log frequency scale and compared to the appropriate target response spectra. The ground motion parameters, namely, computed peak values of PGV/PGA, PGA*PGD/PGV^2, and strong motion duration (time in seconds for Arias intensity to rise from 5 to 75 percent), of the spectrally matched time histories are shown in PSAR Table 3.7-3. Computed cross-correlation coefficients of the two horizontal and one vertical components of the matched time histories to FIRS are shown in PSAR Table 3.7-4.
- Comparison of the scaled response spectra of generated time histories to the target response spectrum are shown in PSAR Figures 3.7-6 through 3.7-8. The acceleration, velocity, displacement, and normalized Arias intensity plots for the matched time histories for three components of each seed ground motion are shown in Figures 3.7-10 through 3.7-24.

The staff finds that the criteria specified to develop design ground motion time histories meet the acceptance criteria in SRP 3.7.1.II.1.B, Option 2, as follows:

- Five sets of design ground motion time histories, each set consisting of three mutually orthogonal directions (two horizontal and one vertical) are used as shown in PSAR Figures 3.7-10 through 3.7-24, which exceeds the minimum of four sets recommended in NUREG/CR-5347 and SRP Section 3.7.1 for linear analysis.
- Each of the three time histories of each set is shown to be statistically independent based on the absolute value of correlation coefficient between the two horizontal and one vertical components in PSAR Table 3.7-4, as amended by CPA Supplement 2 dated October 30,

2025 (TVA 2025-TN13046, TVA 2025-TN13027), being in the range 0.012 to 0.145, and do not exceed the SRP Section 3.7.1 value of 0.16 to be considered statistically independent.

- The strong motion duration (time in seconds for Arias intensity to rise from 5 to 75 percent) shown in the last column of PSAR Table 3.7-3, with a range from 6.16 to 33.72 seconds, exceeds the minimum acceptable value of six seconds.
- The ratios PGA/PGV and PGA*PGD/PGV^2 based on peak values of ground motion acceleration, velocity, and displacement shown in PSAR Table 3.7-3 are generally consistent with typical values for magnitude and distance of earthquakes in Table 3-5 of NUREG/CR-6728.
- The plots in PSAR Figures 3.7-10 through 3.7-24 of the matched time histories for the two horizontal and vertical components for each of the selected earthquake records in PSAR Table 3.7-2 indicate that the acceleration, velocity, and displacements are generally compatible and do not result in displacement baseline drift.
- The plot in PSAR Figure 3.7-9 of the smooth power density of the two horizontal matched time histories of each of the selected five seed records indicates no significant gaps in energy at any frequency over the frequency range of interest (approximately 0.3 to 100 Hz).
- Consistent with Approach 2 of SRP Section 3.7.1, the criteria specified for the modified time histories include time step less than 0.005 seconds, average of the scaled response spectra of the five sets of modified ATHs do not fall more than 10 percent below the target spectrum at any frequency, and the modified ATH response spectra do not exceed the target spectrum by more than 30 percent over the frequency range of interest.

Based on the information provided by the applicant, the staff finds the CRN-1 design acceleration time histories to be acceptable because the response spectra generated from the design time histories satisfy the enveloping criteria prescribed in SRP Section 3.7.1.II.1.B.

3.7.1.3.2 Percentage of Critical Damping Values

The staff reviewed PSAR Section 3.7.1.2 “Percentage of Critical Damping Values,” against the acceptance criteria in SRP Section 3.7.1.II.2, and noted the following:

- PSAR Table 3.7.5, as amended by CPA Supplement 2 dated October 30, 2025 (TVA 2025-TN13046, TVA 2025-TN13027), lists the structural damping values used in the seismic analysis of the IRB structures and components for structural materials including steel-plate composite, welded and bolted steel, and reinforced concrete for both seismic OBE and SSE. The PSAR, as amended by CPA Supplement 2 dated October 30, 2025 (TVA 2025-TN13046, TVA 2025-TN13027), states the structural damping values for steel-plate composite components are from RG 1.61, Revision 2 (NRC 2023-TN13035).
- The PSAR states that the lower OBE damping values are used for generating in-structure demands for seismic qualification of equipment and systems in accordance with the guidance in Section C.1.2 of RG 1.61, Revision 2, and higher SSE damping values are used to develop demands for structural design.
- The PSAR states that soil subgrade damping values are strain-compatible to the free-field strains generated by the design level earthquake and developed based on the site response analysis discussed in PSAR Section 2.5.2 (which the staff evaluated in Section 2.5.2 of this report). The PSAR also states that the strain-compatible damping of subgrade materials is limited to 15 percent, in accordance with SRP Section 3.7.1, RG 1.208 (Appendix E), and ASCE/SEI 4-16.

- The PSAR also notes that damping values for seismic evaluation of subsystems are obtained using procedures in PSAR Section 3.7.3.5 (which is evaluated by staff in Section 3.7.3.5 of this report).

Based on the information provided by the applicant, the staff finds that the proposed critical damping values meet the acceptance criteria in SRP Section 3.7.1.II.2 because of the following:

- The damping values in PSAR Table 3.7-5, as amended, for structural materials are consistent with the values recommended in RG 1.61, Revision 2.
- The use of lower OBE damping values for generating in-structure response spectra for seismic qualification of equipment and systems is consistent with the guidance in Section C.1.2 of RG 1.61, Revision 2.
- The strain-compatible soil damping values developed based on site response analysis in PSAR Section 2.5.2 are evaluated by the staff in Section 2.5.2 of this report. These are less than the upper limit for soil damping of 15 percent recommended in SRP Section 3.7.1.II.2.
- The staff evaluation of damping values for seismic subsystems described in PSAR Section 3.7.3.5 is provided in Section 3.7.3.5 of this report.

3.7.1.3.3 Supporting Media for Seismic Category 1 Structures

The staff reviewed PSAR Section 3.7.1.3, “Supporting Media for Seismic Category I Structures,” against the acceptance criteria in SRP Section 3.7.1.II.3, which states that to be acceptable, the description of supporting media for each seismic Category I structure should include foundation embedment depth, depth of soil over bedrock, soil layering characteristics, design groundwater elevation, dimensions of the structural foundation, total structural height, and soil properties such as shear wave velocity, shear modulus, and material damping, including strain-dependent effect, as well as Poisson’s ratios and density as a function of depth.

Based on review of PSAR Section 3.7.1.3, the following are noted:

- The IRB structures are founded on a common mat foundation deeply embedded in soil and rock below. The PSAR refers to Sections 2.4 and 2.5 for description of the supporting media at the CRN-1 site.
- The foundation’s general arrangement, embedment depth (115.6 ft approx.), and total structural height (223 ft approx.) of the IRB are provided in PSAR Table 1.2-2 and Figure 1.2-3, and foundation dimensions (outer diameter 37 m and thickness 1.22 m) are provided in PSAR Figure 3.8-8.
- To address uncertainties related to determination and variation of subgrade conditions, the SSI analysis results described in PSAR Appendix 3C uses two subgrade cases (Case 1 and Case 2), each with three sets of subgrade profiles consisting of LB, BE, and UB estimates of subgrade material properties to be representative of the as-built conditions at the site with a minimum coefficient of variation of each layer properties of 0.5. The design uses an envelope of results from the SSI analysis of BE, LB, and UB subgrade profiles.
- Case 1 considers the in-situ soils to be left intact, and engineered backfill is added to raise the plant finished grade elevation to 814.5 ft. Case 2 considers the condition where the residual soil and weathered rock are removed down to competent rock and replaced with engineered backfill to the plant grade. Concrete fill dynamic properties in the excavation are provided in PSAR Table 3C-4.

- The development of dynamic subgrade profiles considers the soils located below the nominal ground water table level of EL 814.5 ft to be fully saturated.
- PSAR Figures 3.7-25 and 3.7-26 show the strain-compatible shear wave and compression velocity, and damping ratio profiles, respectively, as a function of depth used for the PSAR Appendix 3C Case 1 seismic analyses. PSAR Figures 3C-5 and 3C-6 show the strain-compatible shear wave and compression velocity, and damping ratio profiles, respectively, as a function of depth used for the Appendix 3C Case 2 seismic analyses. PSAR Figures 3.7-25 and Figure 3C-5 indicate that the BE shear wave velocity exceeds 1000 ft/s at a depth approximately 40 ft below grade.
- PSAR Table 2.5.4-21 provides a summary of BE engineering property values for the subsurface materials in the power block area, which include soil properties such as shear wave and compression wave velocities, Poisson's ratios, shear modulus, elastic modulus, unit weight, and density.

The staff reviewed the description of the supporting media for CRN-1 seismic Category I structures (i.e., IRB) to ensure that the application included sufficient information. The applicant adequately described the supporting media cases for SSI analyses of its seismic Category I structures (i.e., the IRB), including the structural foundation dimensions and depth, subgrade profiles, and the soil properties. The applicant provided tables and figures that show the shear wave velocity; shear modulus; material damping, including the strain-dependent effects; and the density of the soil types. The staff finds that the descriptive information and referenced tables and figures in PSAR Section 3.7.1.3, Appendix 3C, and PSAR Sections 2.4 and 2.5 contain sufficient information on the supporting media for a CPA and are consistent with the acceptance criteria in SRP Section 3.7.1.II.3.

3.7.1.4 Conclusion

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the seismic design parameters for the design of important-to-safety plant SSCs are acceptable and meet the applicable seismic design basis requirements of 10 CFR Part 50, Appendix A, GDC 2, and 10 CFR Part 50, Appendix S, by considering the most severe earthquake recorded for the site with an appropriate margin and considerations for two levels of earthquakes (safe shutdown earthquake (SSE) and operating basis earthquake (OBE), with OBE set as one-third of SSE), appropriate design response spectra and time histories, and appropriate seismic analysis inputs (e.g., damping, supporting media characteristics). This provides reasonable assurance that the seismic design parameters are adequate for use in the seismic analysis and design of the CRN-1 seismic Category I SSCs to withstand design seismic loadings. Therefore, the NRC staff finds that the information provided by the applicant is sufficient for preliminary design, meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section, and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35. The staff will confirm that the final design conforms to these seismic design basis parameters during the evaluation of the CRN-1 FSAR.

3.7.2 Seismic System Analysis

3.7.2.1 Introduction

PSAR Section 3.7.2, "Seismic System Analysis," describes the methods used by the applicant to perform seismic analyses of seismic Category 1 structures and their preliminary results for the CRN-1 seismic design. The PSAR description addresses the following areas:

- seismic analysis method
- natural frequencies and responses
- procedures used for analytical modeling
- soil-structure interaction (SSI) analyses
- in-structure response spectra (ISRS) development
- three components of design ground motion
- combination of modal responses
- interaction of non-seismic Category 1 structures with seismic Category 1 SSCs
- effects of parameter variations on responses
- use of equivalent vertical static factors
- methods used to account for torsional effects
- comparison of responses
- analysis procedure for damping
- determination of seismic overturning moments and sliding forces, structure to soil pressures, and frictional forces for seismic Category 1 structures

Relevant consensus standard ASCE 4-16, "Seismic Analysis of Safety-Related Nuclear Structures"; licensing topical reports NEDO-33914-A, "BWRX-300 Advanced Civil Construction and Design Approach," and NEDC-33926P-A, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design"; and NUREG/CR-7193, *Evaluations of NRC Seismic-Structural Regulations and Regulatory Guidance, and Simulation-Evaluation Tools for Applicability to Small Modular Reactors (SMRs)*, are identified along with relevant NRC guidance in SRP 3.7.2, DC/COL-ISG-017, RG 1.61, RG 1.92, and RG 1.122.

The applicant amended PSAR Section 3.7.2 by CPA Supplement 2 dated October 30, 2025 (ML25275A105); CPA Supplement 4 dated December 2, 2025 (TVA 2025-TN13044); CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029); CPA Supplement 6 dated February 4, 2026 (TVA 2026-TN13031); and CPA Supplement 7 dated March 2, 2026 (TVA 2026-TN13043).

3.7.2.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), requires applicants to address the design bases (in this case the methods used for dynamic modeling and seismic analyses including SSI to determine seismic demands) and the relation of the design bases to the principal design criteria, which in this case is the adequacy of the analysis methods used to meet GDC 2 for protection against earthquake natural phenomena.
- GDC 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions, with consideration of (1) the most severe natural phenomena historically reported for the site and surrounding area, (2) appropriate combinations of the effects of normal and

accident conditions with effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

- 10 CFR Part 50, Appendix S, requires in part that, for safe shutdown earthquake (SSE) ground motions, SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after vibratory ground motion through design, testing, or qualification methods. The evaluation must account for soil-structure interaction (SSI) effects and the expected duration of the vibratory motion. If the operating-basis earthquake (OBE) is set at one-third or less of the SSE, an explicit analysis or design is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied. Appendix S also requires the horizontal component of the SSE ground motion in the free field at the foundation level of the structures to be an appropriate response spectrum with a peak ground acceleration of at least 1/10 the acceleration of gravity (0.1g).

SRP Section 3.7.2, Revision 4, "Seismic System Analysis," issued September 2013, lists acceptance criteria for seismic system analyses to meet the regulatory requirements above and provides review interfaces with SRP Sections 2.5.1 through 2.5.3 for review of geological and seismological information to establish the free-field ground motion, SRP Section 2.5.4 for geotechnical parameters and methods employed in the analysis of free-field soil media and soil properties, SRP Section 3.7.1 for design ground motion, and SRP Sections 3.8.1 through 3.8.5 for design of seismic Category 1 structures for all applicable load combinations.

Based on PSAR Table 1.9-20, the following NRC RGs and other guidance documents also contain relevant guidance for this review:

- RG 1.61, Revision 2, *Damping Values for Seismic Design of Nuclear Power Plants*, issued December 2023, for determining acceptability of damping values used in the dynamic seismic analyses if seismic Category 1 SSCs (ADAMS Accession No. NRC 2023-TN13035).
- RG 1.92, Revision 3, *Combining Modal Responses and Spatial Components in Seismic Response Analysis*, issued October 2012, for guidance concerning acceptable methods for combining seismic modal responses and spatial component response analysis (NRC 2012-TN13038).
- RG 1.122, Revision 1, *Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components* (NRC 1978-TN13039).
- DC/COL-ISG-017, "Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil-Structure Interaction Analyses," issued March 24, 2010 (NRC 2010-TN13036).
- NUREG/CR-7193, *Evaluations of NRC Seismic-Structural Regulations and Regulatory Guidance, and Simulation-Evaluation Tools for Applicability to SMRs*, March 2015 (NRC 2013-TN13042).
- DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications," Interim Staff Guidance, October 2022 (NRC 2022-TN12894).
- NEDO-33914-A, Revision 2, "BWRX-300 Advanced Civil Construction and Design Approach," Licensing Topical Report (GE Hitachi 2022-TN13022).
- NEDC-33926P/NEDO-33926P-A, Revision 4, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building (RB) Structural Design" (proprietary/non-

proprietary), Licensing Topical Report (ADAMS Accession No. ML25351A085 package, ML25351A087 proprietary; GE Hitachi 2025-TN13024 public); NRC Safety Evaluation is included as the front matter of the LTR -A version.

3.7.2.1 Technical Evaluation

The staff reviewed PSAR Section 3.7.2, as amended, addressing seismic system analysis to develop seismic demands for design of seismic Category 1 structures, against the regulatory requirements identified above and the relevant acceptance criteria in SRP Section 3.7.2, Revision 4, and the guidance in RGs and Interim Staff Guidance (ISG) referenced above. Meeting the applicable SRP Section 3.7.2 acceptance criteria provides assurance that seismic Category I structures will be adequately designed to withstand the effects of the SSE and therefore will be able to perform their intended safety function during and following the earthquake. The specific areas of the staff review included seismic analysis methods, natural frequencies and response, analytical modeling procedures, SSI analysis, development of ISRS, three components of design ground motion, combination of spatial and modal responses, interaction between non-seismic Category I and seismic Category I structures, effects of parameter variations on responses, equivalent vertical static factors, consideration of torsional effects, analysis procedure for damping, and determination of seismic overturning moments and sliding forces, structure to soil pressures, and frictional forces.

The sections below present the staff's evaluation of the seismic system analysis for the BWRX-300 design at the CRN-1 site. Section 3.7.1 of this report presents the staff's evaluation of the seismic design parameters, and Section 3.7.3 of this report presents the staff's evaluation of the seismic subsystem analysis.

3.7.2.1.1 Seismic Analysis Method

The staff reviewed PSAR Section 3.7.2.1, "Seismic Analysis Method," as amended by CPA Supplement 7 dated March 2, 2026 (TVA 2026-TN13043), regarding the seismic analysis method(s) used for the CRN-1 design, against the acceptance criteria in SRP Section 3.7.2.II.1. From a review of the PSAR section, the staff noted the following:

- The applicant used the one-step approach in accordance with the guidance in Section 5.0, "Design Analysis," of NRC-approved LTR NEDO-33914-A, Revision 2, "BWRX-300 Advanced Civil Construction and Design Approach" (GE Hitachi 2022-TN13022) for seismic analyses of the seismic Category 1 IRB using three-dimensional (3-D) finite element models (described in PSAR Section 3.7.2.3), which include consideration of the following effects:
  - seismic interaction of the deeply embedded RB structure with surrounding subgrade (SSI)
  - variation in the soil and structural parameters
  - hydrodynamic loads (mass and stiffness) from water in fuel and equipment pools
  - structure-soil-structure interaction (SSSI) through the subgrade with the adjoining power block structures, namely, the RWB, CB, TB, service building, and reactor auxiliary structures
- The one-step approach implements the process laid out in Section 3.1.2 of ASCE/SEI 4-16 by which all seismic responses and demands for all frequencies of interest in a structural system are determined in a single SSI analysis.

- The implementation of the one-step approach includes a few modifications to the guidance in Section 5.0 of NEDO-33914-A aimed to represent realistic site, structures, and construction conditions described in PSAR Sections 3.7.2.3 and 3.7.2.4.
- Linear elastic SSI analysis is performed in the frequency domain using SASSI computer code using structural models that accurately represent the geometry and dynamic properties of the IRB and the subgrade and have a refined finite element mesh that can transmit the entire frequency range of interest. The models assume isotropic elastic material properties of structural members and surrounding subgrade and neglect any non-linearity at the soil-structure interface, which the staff found acceptable in its evaluation of L&C 5 on LTR NEDO-33914-A in Section 2.5.4.4 of this report and verified by staff in Section 3C.3 of this report The solution for seismic response of the IRB is obtained in the frequency domain for a selected set of frequencies and then interpolated for other frequency points.
- The SSI analyses are performed using the sub-structuring method in Section 5.4 of ASCE/SEI 4-16, where the seismic response of the SSI system is obtained by subdividing the problem into a series of simpler subproblems that can be solved separately and the results combined using the principle of superposition based on linear elastic behavior assumption.
- To account for primary non-linearity and uncertainties of subgrade materials, a set (LB, BE, and UB) of profiles of strain-compatible subgrade properties developed in PSAR Section 2.5.2 and described in PSAR Section 3.7.1.3 (and evaluated in Sections 3.7.1.3 and 2.5.2 of this report) are used as input in the SSI analyses.
- Five sets consisting of three orthogonal ground motion ATHs each, described in PSAR Section 3.7.1.1.2 (which staff evaluated in Section 3.7.1.3.1.2 of this report), are used as input motion for the seismic analyses to mitigate uncertainty in the computed responses.
- The input ground motion ATHs are applied as input to the bottom of the IRB foundation as vertically propagating coherent shear waves for horizontal components of input motion and compression waves for the vertical component of input motion.
- Uncertainties related to variations of the input SSI parameters and the wave propagation pattern through the CRN-1 site are addressed by results of sensitivity analyses performed and described in PSAR Section 3.7.2.9. Results of sensitivity analyses are included in the RB seismic design basis when significant exceedances are observed using guidance in Section 5.3 of NEDO-33914-A.
- Design responses are obtained from the analysis of each of the three ground motion components for each of the five sets of ATHs using methodology described in PSAR Sections 3.7.2.5 and 3.7.2.6. The seismic design demands are calculated as the average of the results obtained from the analyses of the five sets of ATHs.

The staff further noted that the PSAR references in Sections 2.4, 2.5, 3.3, 3.7, and 3.8 and Appendices 3B and 3C, and the NRC-approved LTR NEDO-33914-A, Revision 2, “BWRX-300 Advanced Civil Construction and Design Approach” (GE Hitachi 2022-TN13022), as part of the technical basis for the seismic analysis and design of CRN-1 seismic Category I structures. The staff also noted that there are five L&Cs associated with the use of LTR NEDO-33914-A that are addressed in other PSAR sections. The staff’s evaluation of the applicant’s compliance with the five L&Cs associated with LTR NEDO-33914-A is documented and found acceptable in Section 2.5.4.4 of this report.

The staff reviewed the descriptions contained in PSAR Section 3.7.2.1 and found them acceptable because the one-step linear-elastic seismic SSI analysis method used for CRN-1 seismic Category I SSCs is a generally recognized dynamic analysis method in industry standard ASCE/SEI 4-16, is found acceptable by the staff evaluation of L&C 5 on NRC-approved LTR NEDO-331914-A, and is verified in Section 3C.3 of this report, and the information provided is adequate to meet the acceptance criteria for time history dynamic analysis method in SRP Section 3.7.2.II.1.A.

3.7.2.1.2 Frequencies of Analysis

The PSAR states that the frequencies of analyses are determined following the guidance of Section 5.3.2 of LTR NEDO-33914-A, whereby cutoff frequencies are established as the largest value required by the following criteria:

- twice the highest dominant frequency of the coupled soil structure system, or
- the highest structural frequency of interest, or
- the frequency at which the Fourier amplitude of input motion has passed its peak value and has reached 10 percent of the peak value, and
- 20 Hz

The PSAR states that the value of cutoff frequency determined by the above criteria is used for the analysis of the stiffest subgrade profile and model with UB structural stiffness properties. The analyses of softer subgrade profiles may use lower values of cutoff frequency per guidance in Section 5.3.2 of NEDO-33914-A. The frequencies of analysis are selected at sufficiently small frequency intervals, transfer function amplitude results at key locations are inspected to detect any anomalies, which, if any, are evaluated using additional analysis frequencies to assure accuracy of calculated responses.

The staff finds the approach for determining frequencies of analysis and cutoff frequency acceptable because it is based on NRC-approved LTR NEDO-33914-A, Section 5.3.2.

<u>Natural Frequencies and Response Loads</u>

The staff reviewed PSAR Section 3.7.2.2, "Natural Frequencies and Response Loads," against the acceptance criteria in SRP Section 3.7.2.II.2. From review of PSAR Section 3.7.2.2, the staff noted the following:

- The PSAR refers to Appendix 3C for the significant frequencies and SSE responses of the CRN-1 seismic Category 1 structures (i.e., IRB). PSAR Appendix 3C includes the following related information:
  - Section 3C.5 and Table 3C-5 include dominant cut-off frequencies of SSI system response for three subgrade profiles for analysis Cases 1 and 2, which were 35 Hz for LB cases, 51 Hz for BE cases, and 70 Hz for UB cases. The SSI analysis cases are performed for frequencies narrowly spaced starting at 0.012 Hz to respective cutoff frequency for each of the subgrade profiles.
  - Appendix 3C provides a summary of SSI structural responses (element forces, moments, acceleration relative displacements, sample enveloped and broadened in-structure response spectra) at key locations from the Case 1 and Case 2 SSI analyses in the form of tables and figures.

- – Average stress results from the corresponding five sets of ATHs represent the response of the structure for each of the analysis cases in Table 3C-3. Stress results from the analyses of the three subgrade profiles for Case 1 and Case 2 and sensitivity analyses cases are enveloped to provide the design seismic demands that account for uncertainties in and variations of the subgrade properties and seismological conditions at the site.
- The seismic responses of the IRB are characterized using results of the SSI analyses for responses at key structural locations and stress responses of key structural members selected per the guidance in Section 5.3.1 of LTR NEDO-33914-A. The selected key nodal locations and key structural members are shown in PSAR Table 3C-1 and Table 3C-2, respectively.

Based on the above review, the staff finds that the information provided in PSAR Section 3.7.2.2 and Appendix 3C regarding dominant frequency range and responses from the preliminary SSI analyses to be acceptable because the scope and nature of the information provided are consistent with the acceptance criteria in SRP Section 3.7.2.II.2 for the purpose of issuance of a construction permit.

3.7.2.1.3 Procedures Used for Analytical Modeling

The staff reviewed PSAR Section 3.7.2.3, as amended by CPA Supplements 2 and 5, that describes the criteria and procedures used in the analytical modeling of seismic Category 1 structures (i.e., the IRB) for seismic system analysis in accordance with the guidance in SRP Section 3.7.2.II.3.

The staff reviewed the analytical modeling procedures in PSAR Section 3.7.2.3 for the integrated reactor building (IRB) model and noted the following:

- The 3-D finite element (FE) model for SSI analysis is developed following the guidance in SRP Section 3.7.1, SRP Section 3.7.2, and RG 1.61.
- It also meets the structural modeling requirements of ASCE/SEI 4-16, Section 3, and the one-step modeling requirements and guidance in NRC-approved LTR NEDO-33914-A, Section 5.1.1.
- The model consists of the IRB structure, the surrounding grade, the excavated volume of subgrade materials replaced by the embedded portion of the RB structure, and the near field backfill materials.
- Includes construction elements of the CRN-1 excavation plan consisting of concrete mud mat under IRB mat foundation, an annulus of concrete fill extending full embedment depth adjacent to the wall, and a concrete shelf.
- Solid elements are used to model concrete fill. Additional solid elements are added to model temporary excavation support with soil properties for base case and concrete properties for sensitivity analyses.
- The IRB structure is primarily constructed of diaphragm plate steel-plate composite (DP-SC) modules.

The staff finds that IRB structural and subgrade features are adequately modeled and will result in a more realistic seismic response, and acceptable to meet the relevant guidance in the SRP for issuance of a CP.

3.7.2.2.3.1 *Dynamic Finite Element Modeling of Integrated Reactor Building*

The staff reviewed PSAR Section 3.7.2.3.1, as amended by CPA Supplements 2 and 5, regarding dynamic modeling of masses and loads on the IRB and noted the following:

- Detailed information on the IRB models is provided in PSAR Appendix 3B.
- The IRB FE model represents masses expected to be present during an earthquake.
- These include 50 percent of specified live loads, 25 percent of the design snow loads, and inertia associated with hydrodynamic effects of fluids contained in various pools inside the RB.
- Impulsive and convective (sloshing) components of hydrodynamic effects are considered.
- Hydrodynamic mass is included by distributing the horizontal impulsive fluid mass over the pool walls and lumping the vertical fluid mass on the pool bottom.
- Sloshing hydrodynamic effects are considered by applying quasi-static sloshing pressure loads applied on pool wall per Section 9.3 of ASCE/SEI 4-16, and hydrodynamic effects of the breathing mode are considered per Section 9.4 of ASCE/SEI 4-16 (PSAR Section 3.7.2.3.1, as amended by CPA Supplement 2 dated October 30, 2025 (TVA 2025-TN13046, TVA 2025-TN13027).
- Beam and shell elements are used to represent configuration of all main structural members, with DP-SC walls, slabs, and mat foundation modeled using thick shell elements.
- FE model includes gross discontinuities such as large openings and member eccentricity.
- Member eccentricity and offsets are accounted for by rigid beam and shell elements or by adjusting properties of centerline modeled elements.
- Welded connections between DP-SC elements are modeled as rigid.
- Spring elements with appropriate stiffness properties are used to capture interaction at soil-structure interface connecting the RB structural and subgrade FE models, to calculate dynamic earth pressures, and to determine whether separation between RB wall and soil occurs.
- Values of modulus of elasticity and Poisson's ratio representing material stiffness are determined per the design codes.
- Effective thicknesses of DP-SC elements are developed per Section 5.5 of LTR NEDC-33926P and account for stress level and concrete cracking.
- Effects of variation of structural stiffness and damping properties are considered.
- Mesh sizes in FE models are sufficiently refined to be capable of transmitting the entire frequency range of interest. The FE mesh is smaller or equal to one-fifth of the smallest wavelength transmitted through the soil (per ASCE/SEI 4-16).
- Lower boundary of the SSI model is established at a distance deeper than at least two times the depth of RB embedment and at least three times the largest foundation dimension from bottom of slab per SRP Section 3.7.2.II.4.

Based on the above review, the staff finds that the analytical modeling procedures and considerations for developing the IRB model are rational following best practices and consistent with the guidance in SRP Section 3.7.2.II.3 and consensus standard ASCE/SEI 4-16, and therefore acceptable. The staff evaluation of the detailed implementation of the IRB model(s) used for SSI analyses described in PSAR Appendix 3B is provided in Section 3B.3 of this report.

3.7.2.2.3.2 *Dynamic Modeling of Subsystems, Components and Equipment*

The staff reviewed PSAR Section 3.7.3.3.2 regarding dynamic modeling of subsystems and noted the following:

- The dynamic properties of subsystems, components, and equipment are included in the
- integrated RB structural model based on the decoupling criteria of SRP Section 3.7.2.II.3.B as stated in NEDO-33914-A and Section 3.7 of ASCE/SEI 4-16, which depend on the ratios of the mass and first natural frequency of the subsystem, component, or equipment to those of the supporting structure.
- To capture the dynamic coupling effects of the reactor pressure vessel (RPV), the dynamic properties of the RPV and its components are represented by a lumped mass stick (LMS) model capable of capturing the significant modes of the RPV seismic response.
- The RPV LMS model is connected to the RB structural model using local spring elements, representing the stiffness of the RPV support skirt and the horizontal stabilizers.

The staff finds the inclusion of the RPV as a coupled subsystem and modeled as an LMS model in the IRB model acceptable because it is based on the decoupling criteria of SRP Section 3.7.2.II.3.

3.7.2.1.4 Soil-Structure Interaction Analysis

The staff reviewed PSAR Section 3.7.2.4, as amended by CPA Supplement 4 (TVA 2025-TN13045), on the modeling methodology used in the seismic system analysis of CRN-1 seismic Category 1 structures (the IRB) to account for the SSI effects in accordance with guidance in SRP Section 3.7.2.II.4.

From the review of the information in PSAR Section 3.7.2.4, the staff noted the following:

- SSI analyses of the IRB are performed per the methodology in PSAR Section 3.7.2.1 (evaluated by staff in Section 3.7.2.1 of this report) for two bounding cases of subgrade conditions (referred to as Case 1 and Case 2 SSI analyses in PSAR Appendix 3C).
- Case 1 considers a range (LB, BE, UB) of subgrade conditions consisting of engineered backfill to grade level placed on top of existing fill, in-situ residual soil, weathered rock, and in-situ base rock, along with a standalone FE SSI model of only the deeply embedded IRB.
- Case 2 considers a range (LB, BE, UB) of subgrade conditions where the in-situ soil and weathered rock are excavated to competent rock with the FE SSI model of the IRB combined with simplified representative dynamic models of the surrounding power block structures (PBSs; or consisting of RWB, CB, TB, service building, and reactor auxiliary structures). This model also addresses structure-soil-structure interaction (SSSI) effects of the surrounding PBSs on the deeply embedded IRB.
- The consideration of two different free-field site conditions for the SSI analyses (Case 1 and Case 2) is intended to add margin to account for absence of the drilled shafts in the analyzed models due to their low design maturity at the time of the CPA. Case 2 model is considered more representative of the expected plant conditions, and results of the two

cases are enveloped to obtain design seismic demands. For both Case 1 and Case 2, the UB SSI analysis subgrade profiles are appropriately used in conjunction with compatible LB damping values, and vice versa. Nevertheless, the staff finds that the representativeness of the Case 1 and Case 2 soil profile conditions to actual as-constructed soil profile conditions should be verified by the applicant in the FSAR per confirmatory OL action item A-OL-3.7-1 stated further below.

- Sensitivity studies are performed using the Case 2 model to evaluate the effects of parameter variations on the IRB responses, as discussed in PSAR Section 3.7.2.9 (and evaluated in Section 3.7.2.9 of this report).
- Detailed implementation of the linear SSI analyses using SASSI computer code and preliminary results are described in PSAR Appendix 3C.
- Key requirements and approaches considered in the seismic SSI analyses included:
  - Implementation of DC/COL-ISG-017 guidance and limitations related to seismic analysis of deeply embedded structures, intended to ensure the deterministic SSI analyses use ground motion inputs that are hazard consistent with the results of probabilistic site response analysis (SRA) at the foundation bottom and at ground surface. The consistency between the free-field motion at the bottom of the RB foundation (in PSAR Figure 3.7-3) used as input for the deterministic SSI analysis and probabilistic SRA is checked as described in Section 3.7.1.1, using the procedure described in Section 5.3.4.1 of LTR NEDO-33914-A.
  - Coupling of soil and structures: Coarse FE models representing the dynamic properties of the surrounding buildings and foundations are included in the combined Case 2 FE model used for evaluating the SSSI effects of the adjacent power block structures on the RB. These models are sufficiently refined to capture the global modes of vibration of the RWB, CB, TB, service building, and reactor auxiliary structures with significant (>20 percent per Section 5.3.7 of NEDO-33914-A) modal mass participations in the three orthogonal directions.

<u>OL Action Item A-OL-3.7-1</u>: TVA should confirm/verify in the FSAR (OLA) that the subgrade profile conditions (intended to be representative of as-built conditions) considered in the Case 1 and Case 2 models (PSAR Sections 3.7.2.4, 3B.2, 3C.1) for SSI and SSSI analyses are in fact representative or conservatively bounding of actual as-built soil subgrade conditions.

Based on the review and confirmatory OL action item A-OL-3.7-1 above, the staff finds that the applicant's modeling methodology, subgrade conditions, modeling cases considered, use of hazard consistent input motions, and consideration of coupling of soil and structures for SSI and SSSI analyses are reasonable, conservatively biased, and consistent with the guidance in SRP Section 3.7.2.II.4, DC/COL-ISG-017, and NRC-approved LTR NEDO-33914-A, and therefore acceptable. The staff's evaluation of the detailed implementation and results of the IRB SSI analyses described in PSAR Appendix 3C is provided in Section 3C.3 of this report.

3.7.2.1.5 Development of In-Structure Responses

The staff reviewed the procedures and methods in PSAR Section 3.7.2.5 used in developing ISRS and ATHs, in accordance with SRP Section 3.7.2.II.5, RG 1.122, and consensus standard ASCE/SEI 4-16 referenced by the applicant. These documents provide guidance and criteria for methods acceptable to the staff for developing two horizontal and vertical ISRS from the response time histories and in-structure ATHs.

3.7.2.2.5.1 *In-Structure Response Spectra (ISRS)*

The staff reviewed PSAR Section 3.7.2.5.1 and noted the applicant stated the following:

- ISRS are developed following the guidance in SRP Section 3.7.2.II.5, RG 1.122, and Section 6.2 of ASCE/SEI 4-16.
- ISRS are developed for required damping levels at different locations within the RB in two horizontal and the vertical directions.
- ISRS are developed from the calculated nodal in-structure responses in accordance with the requirements of Sections 6.2.1(a) and (b) of ASCE/SEI 4-16 with appropriate combination of co-directional responses due to three orthogonal components of input motion.
- ISRS are developed at small frequency intervals and calculated at 301 frequency points equally spaced in the frequency range 0.1 to 100 Hz.
- ISRS are calculated as an envelope of the SSI analysis results of all subgrade profiles.
- ISRS are increased in accordance with Section 5.3 of LTR NEDO-33914-A to address parameter variations.
- Peaks of the enveloping acceleration response spectra (ARS) are broadened ±15 percent in accordance with RG 1.122 to address uncertainties.

Based on the above review, the staff finds that the applicant's process for the development of the ISRS from time histories and accounting for parameter variations conform to the guidance in RG 1.122, SRP Section 3.7.2.II.5, ASCE/SEI 4-16, and NRC-approved LTR NEDO-33914-A, and are therefore acceptable.

3.7.2.2.5.2 In-Structure Acceleration Time Histories (ATHs)

The staff reviewed PSAR Section 3.7.2.5.2 and noted that the PSAR states that, in accordance with the requirements of Section 6.3 of ASCE/SEI 4-16, time histories used in the analysis of subsystems are obtained either

- directly from the results of the SSI analysis as time histories of nodal responses at reference locations of subsystem support locations; or
- by generating synthetic time histories compatible to multi-damping ISRS developed as described in PSAR Section 3.7.2.5.1.

The staff finds the approach for developing in-structure ATHs acceptable because it is consistent with the provisions of recognized consensus standard ASCE/SEI 4-16.

3.7.2.2.5.3 *Relative Displacement*

The staff reviewed PSAR Section 3.7.2.5.3 and noted that relative displacements between different support points of subsystems are evaluated using the support displacement time histories, and maximum design relative displacements are calculated as the envelope of maximum relative displacements obtained for each SSI analysis case. The staff finds this approach to evaluate relative support displacements reasonable and acceptable.

3.7.2.1.6 Three Components of Design Ground Motion

The staff reviewed the method(s) described in PSAR Section 3.7.2.6 for combining the responses from the three components of earthquake ground motion in accordance with the guidance in SRP Section 3.7.2.II.6. The SRP references RG 1.92 for methods acceptable to the staff for combining three spatial components of seismic responses.

From review of PSAR Section 3.7.2.6, the staff noted the following:

- The SSI analyses are performed separately for each of the three directional components of input ground motion using five sets of ATHs. For each set of ATHs used, the seismic response parameters obtained from the analysis of each of the three ground motion components are combined to get the total co-directional response using either of the following three methods permitted in RG 1.92, Revision 3, and Section 4.2.2 of ASCE/SEI 4-16:
  - Time histories of responses due to the three earthquake components are combined algebraically at each time step, and maximum combined response used; alternately, the more conservative absolute sum method in the time domain may be used.
  - The maximum co-directional responses are combined using the 100-40-40 method.
  - The maximum responses due to the three earthquake components are combined using the square-root-of-the-sum-of the-square (SRSS) method.

The staff finds that the applicant’s three proposed methods for combining the three spatial components of seismic responses using the step-by-step time history algebraic or absolute sum method, or the 100-40-40 method, or the SRSS method are in conformance with the guidance in RG 1.92 and recognized consensus standard ASCE/SEI 4-16, and therefore acceptable.

3.7.2.1.7 Combination of Modal Responses

SRP Section 3.7.2.II.7 provides guidance for the combination of modal responses, including consideration of closely spaced modes and high-frequency modes, when using the response spectrum method or the modal superposition time history method of analysis to determine the dynamic response of damped linear systems.

PSAR Section 3.7.2.7 states that modal combination is not utilized for the CRN-1 seismic SSI analysis of seismic Category 1 structures since the analysis is performed using ACS SASSI computer code, which utilizes time history analysis in the frequency domain analysis method, in which the equations of motion are solved for a selected range of frequencies. The staff finds that, since the applicant does not use a method based on modal combination, no further review of the combination of modal responses is needed.

3.7.2.1.8 Interaction of Non-seismic Category I Structures with Seismic Category I SSCs

The staff reviewed the methods described in PSAR Section 3.7.2.8 used to assess non-seismic Category I structures to determine whether their failure under SSE conditions could impair the integrity of seismic Category I SSCs, or result in incapacitating injury to control room occupants, in accordance with the guidance in SRP Section 3.7.2.II.8. Criterion C in SRP Section 3.7.2.8 states that the non-seismic Category 1 structure is analyzed and designed to prevent failure under SSE conditions.

From a review of PSAR Section 3.7.2.8, the staff noted the following:

- To meet the interaction guidance of SRP Section 3.7.2.II.8, Criterion C, the applicant will perform evaluations based on seismic responses of the RWB, CB, TB, service building, and reactor auxiliary structures obtained from the one-step SSSI analyses that incorporate the dynamic response of the IRB and surrounding power block structures, following the approach in Section 6.2 of NRC-approved LTR NEDO-33914-A.
- The seismic interaction evaluations consider limited permanent deformation (Limit State LS-C) structural response to calculate SSE demands on the main lateral load resisting members following Section 6.2 of LTR NEDO-33914-A.
- The resulting displacements and gaps between the RB and adjacent structures are evaluated per Section 6.2 of LTR NEDO-33914-A to ensure no physical interaction between the RB and surrounding structures.

The staff finds that the applicant’s above approach to evaluate seismic interaction between the surrounding non-seismic Category 1 power block structures and the seismic Category 1 RB is acceptable to satisfy Criterion C of SRP Section II.8 because it is based on the guidance in Section 6.2 of NRC-approved LTR NEDO-33914-A.

3.7.2.1.9 Effects of Parameter Variations on Responses

PSAR Section 3.7.2.9 describes the applicant’s considerations to account for the effects of parameter variations on the seismic responses. The staff reviewed PSAR Section 3.7.2.9, as amended by CPA Supplement 6 (TVA 2026-TN13031), and noted the following:

- Uncertainties in structural properties, damping values, subgrade properties, and variations of SSI parameters are accounted for in the analysis and design of BWRX-300 seismic Category I structures by performing sensitivity studies in accordance with SRP Section 3.7.2.II.4, Section 5.1 of ASCE/SEI 4-16, and following the guidance of Section 5.3 of NEDO-33914-A (with some modifications).
- The applicant performed the following sensitivity analyses using the SSI model with the adjacent power block structures. Evaluations of effects of parameter variations on floor responses were based on comparisons of in-structure responses and stress demands at key locations obtained from the sensitivity analyses. If the comparisons show significant exceedances (>10 percent) in the RB seismic response, the results of the sensitivity analysis performed are included in the RB seismic design basis.
  - effects of variation of structural stiffness and damping properties in accordance with Section 5.3.5 of LTR NEDO-33914-A
  - excavation support, backfill, and friction effects in accordance with Section 5.3.8 of LTR NEDO-33914-A
  - groundwater variation effects in accordance with Section 5.3.10 of LTR NEDO-33914-A
  - soil separation effects in accordance with Section 5.3.9 of LTR NEDO-33914-A
  - effects of non-vertically propagating seismic waves in accordance with Section 5.3.3 of LTR NEDO-33914-A
  - non-linear sensitivity evaluation, which concluded the effects of non-linearity are insignificant, and non-linear SSI analysis described in NEDO-33914-A is not required

(L&C 8.5 on NEDO-33914-A does not apply as evaluated in Section 2.5.4.4 of this report)

  – concrete fill strength

- The results of the above sensitivity analyses performed are described in PSAR Appendix 3C.3. The comparisons show that seismic forces and moments do not differ significantly from the base case with some exceptions, notably for the no-friction and cracked concrete sensitivity cases.

Based on the above review, the staff finds the applicant has adequately addressed the effects of parameter variations on seismic responses through sensitivity studies performed in accordance with NRC-approved LTR NEDO-33914-A, consistent with the guidance in SRP Section 3.7.2.II.4 and Section 5.1 of ASCE/SEI 4-16, and therefore acceptable.

3.7.2.1.10 Use of Constant Vertical Static Factors

SRP Section 3.7.2.II.10 allows the use of equivalent static load factors to calculate vertical response loads for the seismic design of nuclear structures if the structure can be demonstrated to be rigid in the vertical direction. However, PSAR Section 3.7.2.10 indicates that constant vertical static factors are not used in the design of the CRN-1 seismic Category I structures; instead, the vertical seismic loads are directly obtained from the SSI analyses using the one-step design analysis approach described in PSAR Section 3.7.2.1. Since the applicant did not use constant vertical static factors, no further staff technical review of this area is needed.

3.7.2.1.11 Methods Used to Account for Torsional Effects

The staff reviewed PSAR Section 3.7.2.3.11 regarding the methods the applicant used to account for torsional effects against the acceptance criteria in SRP Section 3.7.2.II.11. The SRP section states that an acceptable method to account for torsional effects in the seismic analysis of seismic Category I structures is to perform a dynamic analysis that incorporates the torsional degrees of freedom. The SRP section also states that to account for accidental torsion, an additional eccentricity of ±5 percent of the maximum building dimension should be assumed for both horizontal directions.

From the review of PSAR Section 3.7.2.11, the staff noted applicant stated the following:

- Modeling considerations of the IRB include representing the actual locations of centers of masses and centers of rigidity of structural elements to account for actual torsional effects (discussed in PSAR Section 3.7.2.3.1).
- In accordance with the general requirements in Section 3.1 of ASCE/SEI 4-16, the seismic design of the IRB considers accidental torsion to address the effects of non-vertically propagating waves, rotational components of ground motion, and distributions of mass and stiffness in the structure that differ from those represented in the 3-D finite element model used for the seismic response analysis.
- The accidental torsional moment demands are calculated at each floor level as the product of the story shear and 5 percent of the floor plan dimension perpendicular to the story shear direction. Alternatively, the horizontal shear force demands on walls may conservatively be increased by 5 percent to account for accidental torsion.

Based on the review above, the staff finds that the applicant's methodology to account for actual and accidental torsional effects acceptable because it is consistent with the acceptance criteria in SRP Section 3.7.2.II.11, and the alternative methodology included for accidental torsion conservatively meets the acceptance criteria of SRP Section 3.7.2.II.11.

3.7.2.1.12 Comparison of Responses

SRP Section 3.7.2.II.12 states that if both the time history analysis method and the response spectrum analysis methods are used to analyze an SSC, the peak responses obtained from these two methods should be compared to demonstrate approximate equivalency between the two methods. However, PSAR Section 3.7.2.12 states that, because only the time history method is used for the dynamic analysis of seismic Category 1 structures for CRN-1 and the response spectrum method is not used, a direct comparison of responses from the two methods is not applicable. This is acceptable to the staff, and no further technical review of this area is needed.

3.7.2.1.13 Analysis Procedure for Damping

The staff reviewed the applicant's analysis procedure in PSAR Section 3.7.2.13 for damping in accordance with SRP Section 3.7.2.II.13. The guidance in SRP Section 3.7.2.II.13 states that either the composite modal damping approach or the modal synthesis technique can be used to account for element-associated damping.

PSAR Section 3.7.2.13 refers to Section 3.7.1.2 for damping values used in the SSI analysis of seismic Category 1 structures. Refer to Section 3.7.1.3.2 of this report for the staff evaluation of structural material and soil damping values used in the CRN-1 seismic SSI analysis.

In PSAR Section 3.7.1.2, the applicant stated that the damping values in RG 1.61, Revision 2, are used for the dynamic analysis of the seismic Category I SSCs and, for soil and rock materials, the damping values are obtained based on the strain-compatible soil properties generated for each soil profile based on site response analysis and are limited to not exceed 15 percent. In PSAR Sections 3.7.2.13 and 3.7.2.1, both as amended in Rev 1 of the PSAR, the applicant stated that the SSI analysis of the IRB is a frequency domain analysis, which the staff noted is one of the methods provided in SRP Section 3.7.2.II.13(B). The applicant further stated that material damping is introduced using a complex stiffness matrix with complex elastic moduli. For each element, the real part of the elastic modulus represents stiffness, and the imaginary part represents damping, which leads to effective damping ratios that are frequency independent. The staff notes that using a complex stiffness matrix with complex moduli (i.e., modulus of elasticity and shear modulus) is a standard and effective method used in computer codes such as SASSI to represent hysteretic damping in SSI analysis, especially in the frequency domain, and allows for accurate modeling of material damping in structures and soil layers for realistic seismic response prediction.

Based on the above, the staff finds that the applicant's analysis procedure for implementing material damping for SSI analysis in SASSI using the complex stiffness matrix is adequate to meet SRP Section 3.7.2.II.13 acceptance criteria because the linear SSI analysis is implemented in SASSI in the frequency domain, which is among the methods allowed in SRP Section 3.7.2.II.13(B), and as indicated in Section 3.5.1.2 of ASCE 4-16, the complex stiffness matrix approach (which includes both stiffness and damping terms) is used in frequency domain analyses.

3.7.2.1.14 Determination of Seismic Overturning Moments and Sliding Forces, Structure to Soil Pressures, and Frictional Forces for Seismic Category I Structures

The staff reviewed PSAR Section 3.7.2.3.14 against the acceptance criteria in SRP Section 3.7.2.II.14, which provides guidance on the determination of design seismic overturning moments and sliding forces, structure-to-soil pressures beneath the foundation and alongside walls, and soil frictional forces for seismic Category I structures.

In PSAR Section 3.7.2.14, the applicant stated the following:

- Consistent with the guidance of SRP Section 3.7.2.II.14, contact spring elements installed in the one-step SSI models at interfaces between the structure and the subgrade are used for calculation of dynamic bearing pressure demands, seismic driving forces, and overturning moments on the seismic Category I common mat foundation.
- Time histories of the horizontal and vertical seismic forces in the three directions are used.
- The seismic bearing pressure demands under the RB are also calculated in the time domain as the average of five sets of ATHs.

Based on the review, the staff finds that the applicant's approach for determination of design seismic overturning moments and sliding forces, structure-to-soil pressures beneath the foundation and alongside walls, and soil frictional forces for seismic Category I structures is consistent with the guidance in SRP Section 3.7.2.II.14 and therefore acceptable.

3.7.2.2 Conclusion

The staff has reviewed the information provided in the PSAR, as amended, and, for the reasons given above, concludes that the applicant's plant design seismic system modeling and analyses procedures, including soil-structure interaction (SSI) are consistent with relevant portions of SRP Section 3.7.2 and thus are acceptable and meet the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, and 10 CFR Part 50, Appendix S, with respect to the analyses methodology to demonstrate capability of the seismic Category 1 structures to withstand the effects of design-basis earthquakes. The staff review finds that the seismic design analysis procedures reflect appropriate consideration of the most severe earthquake recorded for the site with an appropriate margin consistent with the previously established seismic design parameters, consideration of two levels of earthquakes (with OBE set at one-third SSE), appropriate combination of the effects of normal and accident conditions with the effects of the seismic natural phenomena, the importance of the safety functions to be performed, and the use of a suitable dynamic analysis or suitable qualification tests to develop design demands to demonstrate that SSCs can withstand the seismic and other concurrent loads. The staff finds that the applicant's seismic analysis procedures and descriptions adequately addressed the following considerations: (1) seismic analysis methods; (2) natural frequencies and response loads, basis for selection of frequencies, and number of earthquake cycles; (3) analytical modeling procedures; (4) SSI; (5) in-structure response spectra development; (6) three components of design ground motion; (7) combinations of modal responses; (8) interaction of non-category 1 with category 1 structures; (9) effects of parameter variations; (10) equivalent vertical static factors; (11) methods to account for torsional effects; (12) analysis procedure accounting for damping; and (13) seismic overturning and sliding. Therefore, the NRC staff finds that the information provided by the applicant is sufficient for preliminary design, meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section, and adequately supports the issuance of a CP pursuant to 10 CFR 50.35. The staff will confirm that the final design conforms to these seismic analysis methods and procedures during the evaluation of the CRN-1 FSAR.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Introduction

PSAR Section 3.7.3, "Seismic Subsystem Analysis," describes the methods used by the applicant to perform seismic qualification by analysis of seismic Category 1 subsystems consisting of

- distribution systems, including piping and supports; cable trays and supports; heating, ventilation, and air conditioning ductwork and supports; conduits and supports
- equipment supports
- intervening structural elements between distribution systems and equipment supports and the RB structure
- RPV and internals

The PSAR states input motions for the qualification of these subsystems are in the form of floor response spectra or acceleration time histories from the integrated RB SSI analysis discussed in Section 3.7.2. The PSAR description addresses the following applicable review areas:

- seismic analysis methods
- determination of number of earthquake cycles
- procedures used for analytical modeling
- basis for selection of frequencies
- analysis procedure for damping
- three components of design ground motion
- combination of modal responses
- interaction of non-seismic Category 1 subsystems with seismic Category 1 SSCs
- multiple-supported equipment and components with distinct inputs
- use of equivalent vertical static factors
- torsional effects of eccentric masses
- effects of differential building movements

Relevant consensus standards ASCE 4-16, ASCE 43-19, and ASME BPVC, Section III, Division 1 are identified along with relevant NRC guidance in SRP Sections 3.7.3 and 3.7.2, RG 1.61, RG 1.92, and RG 1.122.

The applicant amended PSAR Section 3.7.2 by CPA Supplement 2 dated October 30, 2025 (ML25275A105) and CPA Supplement 4 dated December 2, 2025 (TVA 2025-TN13044).

3.7.3.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), requires applicants to address the design bases (in this case the methods used for dynamic modeling and seismic subsystem analyses to determine seismic demands) and the relation of the design bases to the principal design criteria, which in this case, the adequacy of the analysis methods used to meet GDC 2 for protection against earthquake natural phenomena.
- GDC 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions, with consideration of (1) the most severe natural phenomena historically reported for the site and surrounding area, (2) appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena, and (3) the importance of the safety functions to be performed.
- 10 CFR Part 50, Appendix S, requires in part that, for safe shutdown earthquake (SSE) ground motions, SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after vibratory ground motion through design, testing, or qualification methods. The evaluation must account for soil-structure interaction (SSI) effects and the expected duration of the vibratory motion. If the operating-basis earthquake (OBE) is set at one-third or less of the SSE, an explicit analysis or design is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied. Appendix S also requires the horizontal component of the SSE ground motion in the free field at the foundation level of the structures to be an appropriate response spectrum with a peak ground acceleration of at least 1/10 the acceleration of gravity (0.1g).

SRP Section 3.7.3, Revision 4, "Seismic Subsystem Analysis Review Responsibilities," issued September 2013, lists acceptance criteria for seismic subsystem analyses (by primarily referring to applicable sections of SRP Section 3.7.2) adequate to meet the relevant requirements above, and provides review interfaces with SRP Sections 2.5.1 through 2.5.3 for review of geological and seismological information to establish the free-field ground motion, SRP Section 2.5.4 for geotechnical parameters and methods employed in the analysis of free-field soil media and soil properties, SRP Section 3.7.1 for design ground motion, SRP Section 3.7.2 for seismic system analysis, and SRP Sections 3.8.1 through 3.8.5 for design of seismic Category 1 structures for all applicable load combinations.

Based on PSAR Table 1.9-20, the following NRC RGs also contain relevant guidance for this review:

- RG 1.61, Revision 2, *Damping Values for Seismic Design of Nuclear Power Plants*, issued December 2023, for determining acceptability of damping values used in the dynamic seismic analyses of seismic Category 1 SSCs (NRC 2023-TN13035)
- RG 1.92, Revision 3, *Combining Modal Responses and Spatial Components in Seismic Response Analysis*, issued October 2012, for guidance concerning acceptable methods for combining seismic modal responses and spatial component response analysis (NRC 2012-TN13038)
- RG 1.122, Revision 1, *Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components* (NRC 1978-TN13039)
- NEDO-33914-A, Revision 2, "BWRX-300 Advanced Civil Construction and Design Approach," Licensing Topical Report (GE Hitachi 2022-TN13022).

3.7.3.3 Technical Evaluation

The staff reviewed PSAR Section 3.7.3, as amended by CPA Supplements 2 and 4, regarding analysis methods of seismic qualification of Category 1 subsystems, against the regulatory requirements identified above and the relevant acceptance criteria in SRP Section 3.7.3, Revision 4. The staff review included analysis methods of seismic Category 1 subsystems consisting of distribution systems (i.e., piping and supports; cable trays and supports; heating, ventilation, and air conditioning ductwork and supports; conduits and supports), equipment supports, intervening structural elements between distribution systems, equipment supports and the RB structure, and the RPV and internals. The staff noted that input motions for seismic analysis of these subsystems are in the form of floor response spectra or ATHs obtained from the seismic SSI analysis of the IRB described in PSAR Section 3.7.2 and evaluated in subsections within Section 3.7.2 of this report. The staff evaluation of the different review areas (similar to Section 3.7.2 of this report) for subsystems is described below.

3.7.3.3.1 Seismic Analysis Methods

The staff reviewed PSAR Section 3.7.3.1, "Seismic Analysis Methods," for subsystems against the acceptance criteria in SRP Section 3.7.2.II.1. PSAR Section 3.7.3.1 states that seismic analyses of CRN-1 subsystems are performed using one of the following methods as described in SRP Section 3.7.2.II.1:

- time history analysis
- response spectrum analysis
- static coefficient method

3.7.3.3.1.1 *Time History Analysis*

The staff reviewed PSAR Section 3.7.3.1.1, "Time History Analysis," and noted the following:

- The method consists of solving the dynamic equations of equilibrium of a lumped mass, distributed stiffness (assuming velocity proportional damping) in the time domain by modal superposition or direct integration, which are summarized.
- The direct integration method is recommended for systems subject to short duration high frequency excitation due to lesser computational effort.
- The numerical integration time step for the solution is selected to be sufficiently small to accurately define the dynamic excitation and render stability and convergence of the solution up to the highest frequency of significance.

The staff finds the time history analysis approaches acceptable because they are recommended methods for dynamic seismic analysis in SRP Section 3.7.2.II.1.A and recognized consensus standard ASCE/SEI 4-16 and have been used in past nuclear power reactor licensing applications.

3.7.3.3.1.2 *Response Spectrum Analysis*

The staff reviewed PSAR Section 3.7.3.1.2, "Response Spectrum Analysis," and noted the following:

- The response spectrum method is a frequency domain modal superposition analysis in which only the peak values of the solution of the decoupled modal equations are obtained and used if only peak dynamic responses are required.
- The advantage of the method is that, for a given input motion, the variables under the integral are only the damping factor and frequency, and the curve constructed giving the maximum values of response as a function of frequency is called the response spectrum for the given input motion and damping.
- For subsystem analysis, the input floor response spectra, obtained from time history analysis of the primary system, is broadened ±15 percent to account for modeling uncertainties.
- Using the calculated natural frequencies of vibration of the subsystems, the maximum values of modal responses are determined directly from the response spectrum and then combined.

The staff finds the response spectrum analysis approaches acceptable because they are recommended frequency domain methods for dynamic seismic analysis in SRP Section 3.7.2.II.1.A and recognized consensus standard ASCE/SEI 4-16 and have been extensively used in past nuclear power reactor licensing applications.

3.7.3.3.1.3 *Static Coefficient Method*

The staff reviewed PSAR Section 3.7.3.1.3, “Static Coefficient,” against the acceptance criteria in SRP Section 3.7.3.II.1, which incorporates by reference the criteria in SRP Section 3.7.2.II.1.B, “Equivalent Static Load Method,” and noted the following:

- The static coefficient method (same as equivalent static load method) may be used for analysis of certain equipment in lieu of dynamic analysis.
- Equivalent static seismic response loads are determined by multiplying the equipment mass by a static coefficient equal to 1.5 times the maximum peak spectral acceleration. A factor less than 1.5 may be used if adequate justification is provided, for example, if the equipment is simple enough such that it behaves essentially as a single-degree-of-freedom system model and the fundamental frequency is greater than the seismic excitation frequency (i.e., frequency corresponding to peak spectral acceleration).
- The method is applicable without further justification only to equipment corresponding to a simple column, beam, or frame type structure supported at a single point.
- If the fundamental frequency of the equipment is between the seismic cutoff frequency and the zero point acceleration (ZPA), the static coefficient may be taken as 1.5 times the peak spectral acceleration between the cutoff and ZPA frequencies.

Based on the above review, the staff finds that the static coefficient seismic analysis approach described in the PSAR is acceptable because it only applies to equipment that can be realistically represented by a simple model and is consistent with the acceptance criteria in SRP Section 3.7.2.II.1.B.

3.7.3.3.1.4 *Seismic Analysis of Piping Systems*

The staff reviewed PSAR Section 3.7.3.1.4, "Seismic Analysis of Piping Systems," regarding methods used for piping seismic analysis against criteria in SRP Section 3.7.2.II.1. From review of the PSAR section, the staff noted the following:

- The time history and response spectrum methods are utilized to obtain dynamic response, as required.
- The procedure for multi-support excitation described in PSAR Section 3.7.3.9 (staff evaluation in Section 3.7.3.3.9 of this report) is followed for both methods. When response spectrum method is used for multi-support piping system, all multi-support response spectra components are simultaneously applied to each piping model and load case.
- Secondary stresses due to anchor displacements are conservatively accounted for in the analysis.
- The dynamic loads on pipe mounted pumps and valves are determined by including pumps and valves in the dynamic analysis model of the piping system.

Based on the above review, the staff finds that the seismic analysis methods and considerations for piping systems described in the PSAR are acceptable because the considerations are appropriate and the dynamic analysis methods are consistent with the methods recommended in SRP Section 3.7.2.II.1.A and evaluated in Sections 3.7.3.3.1.1 and 3.7.3.3.1.2 of this report.

3.7.3.3.1.5 *Seismic Analysis of Equipment*

The staff reviewed PSAR Section 3.7.3.1.5, "Seismic Analysis of Equipment," regarding methods used for seismic analysis of equipment against the acceptance criteria in SRP Section 3.7.2.II.1. From review of the PSAR section, the staff noted the following:

- The time history and response spectrum methods are utilized, as required.
- When the equipment is supported at two or more location points at different building elevations, the response spectra for the most severe attachment point is used. Alternatively, the multi-support excitation procedure described in PSAR Section 3.7.3.9 (staff evaluation in Section 3.7.3.3.9 of this report) is used.
- The relative support displacement determined from dynamic analysis is used for static analysis to determine additional stresses due to differential support displacements.
- Static coefficient method discussed in PSAR Section 3.7.3.1.3 may be used for rigid equipment and frames supported at a single location.

Based on the above review, the staff finds that the seismic analysis methods and considerations for equipment described in the PSAR are acceptable because the considerations are appropriate and the dynamic or static analysis methods used are consistent with the methods recommended in SRP Section 3.7.2.II.1 and evaluated in Sections 3.7.3.3.1.1, 3.7.3.3.1.2, and 3.7.3.3.1.3 of this report.

3.7.3.3.1.6 *Seismic Analysis of Reactor Pressure Vessel (RPV) and Internals*

The staff reviewed PSAR Section 3.7.3.1.6, "Seismic Analysis of Reactor Pressure Vessel (RPV) and Internals," regarding methods used for seismic analysis of RPV and internals against the acceptance criteria in SRP Section 3.7.2.II.1. From review of the PSAR section, the staff noted the following:

- A lumped mass stick (LMS) model representing the dynamic properties of the RPV is included in the IRB model (discussed in PSAR Sections 3.7.2.3.2 and 3.7.3.3.2 to capture the interaction of the IRB structures with the RPV and internals.
- The dynamic responses required as input for the design of the RPV and internals are generated directly from the dynamic analysis of the IRB at the IRB/RPV interface locations.
- Seismic ATHs obtained above are applied to the RPV and internals FE model simultaneously in the three orthogonal directions. Alternatively, two analyses may be performed to calculate the horizontal and vertical RPV responses separately.
- Seismic evaluation of RPV internal components in the horizontal direction is performed using a two-step analysis approach, where seismic ATHs at RPV support location are applied to more detailed FE model of the RPV and internals.
  - The first step consists of obtaining ATHs developed (PSAR Section 3.7.2.5) at the RPV/IRB interface locations from IRB SSI analysis (PSAR Section 3,.7.2).
  - The second step is a multi-support excitation time history analysis of RPV and internals subject to the ATHs generated in the first step using procedure in PSAR Section 3.7.3.9.
  - Time step of the input ATHs is varied to address modeling uncertainties in accordance with requirements in Section 6.3.2 of ASCE/SEI 4-16.

Based on the above review, the staff finds that the seismic analysis methods and considerations for the RPV and internals using a two-step dynamic time history analyses are acceptable because the considerations are appropriate and the dynamic analysis methods used are consistent with the methods recommended in SRP Section 3.7.2.II.1 and consensus standard ASCE/SEI 4-16.

3.7.3.3.2 Determination of the Number of Earthquake Cycles

The staff reviewed PSAR Section 3.7.3.2, “Determination of Number of Earthquake Cycles,” against the acceptance criteria in SRP Section 3.7.3.II.2 and noted the PSAR states that the number of earthquake cycles for subsystem analysis is in accordance with SRP Section 3.7.3.II.2.

The staff finds the methods used for determination of number of earthquake cycles acceptable because the PSAR incorporates by reference in its entirety the related acceptance criteria in SRP Section 3.7.3 paragraph II.2, which also includes consideration of OBE for fatigue analyses for CRN-1, where the OBE is set at one-third of SSE. For this case, the SRP criteria recommend two SSE events with 10 maximum stress cycles (20 full cycles of maximum SSE stress range in total), which is considered equivalent to the cyclic load basis of one SSE and five OBEs with a minimum of 10 maximum stress cycles per earthquake. SRP Section 3.7.3.II.2 also allows an alternative method in which the number of fractional vibratory cycles equivalent to 20 full SSE vibratory cycles may be used (but with an amplitude not less than one third of the maximum SSE amplitude) when derived in accordance with Appendix D to Institute of Electrical and Electronics Engineers (IEEE) Standard 344-2004, “IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Plants,” dated June 8, 2005.

3.7.3.3.3 Procedures Used for Analytical Modeling

The staff reviewed PSAR Section 3.7.3.3, “Procedures Used for Analytical Modeling,” of seismic Category 1 subsystems against the acceptance criteria in SRP Section 3.7.3.II.3, which refers to SRP Section 3.7.2.II.3. From review of the PSAR section, the staff noted that following:

- Mathematical models for dynamic analyses of seismic Category 1 subsystems are developed to reflect dynamic characteristics of the components by discretizing them into a series of interconnected beam elements or other finite elements and lumped masses.
- The node points are selected to coincide with locations of large masses or at points corresponding to significant change in geometry.
- The number of mass node points in the model is considered sufficient if additional node points do not result in more than 10 percent increase in responses in the frequency range of interest (below cutoff frequency).
- The element mesh size is selected on the basis that further refinement has only a negligible effect on the solution results (PSAR Section 3.7.3.3, as amended by CPA Supplement 2, dated October 30, 2025 (TVA 2025-TN13046, TVA 2025-TN13027).
- Modeling of seismic Category 1 subsystems identified, namely, piping systems, equipment, and reactor pressure vessel (RPV) and internals, is described.
- Modeling of piping systems:
  - Modeled as an assemblage of pipe elements with pipe and fluid masses lumped at nodes and concentrated weights modeled as lumped mass rigid systems.
  - Torsional effects of valve operators and equipment with offset are explicitly included in the model.
  - Pipe length between mass points is limited to no greater than the length with a fundamental frequency equal to the cutoff frequency.
  - Pipe guides, snubbers, and supports are modeled with representative stiffness, unless shown to be rigid.
- Modeling of equipment:
  - Represented by lumped mass systems consisting of discrete masses connected by weightless beam elements.
  - Number of masses chosen to include all significant modes and lumped where significant weight is located.
  - For equipment with free-end flexible overhang, a mass is lumped at the overhang span.
  - When mass is lumped between two supports, it is conservatively located to maximize displacement and lower natural frequency.
- Modeling of RPV and internals:
  - Because of significant dynamic interaction between RB and RPV, the RPV is integrated into the IRB model as a lumped mass stick (LMS) model.
  - The mathematical model of RPV and internals consists of a linear elastic LMS model and detailed 3-D FE model with appropriate stiffnesses.
  - Dynamic analysis of RPV is performed by a 2-step process.

- Dynamic response of selected RPV and internal components is determined from a subsystem analysis after the system response is found from the IRB model.
- Hydrodynamic effects of water enclosed are accounted for.
- Fuel assembly is adequately modeled.
- Asymmetric equipment is modeled using FE or LMS methods.

Based on the above review, the staff finds that the applicant has appropriately identified seismic subsystems, and the analytical modeling procedures for dynamic analysis of the subsystems follow modeling best practices, are consistent with the guidance in SRP Section 3.7.2.II.3, and are therefore acceptable.

3.7.3.3.4 Basis for Selection of Frequencies

The staff reviewed PSAR Section 3.7.3.4, "Basis for Selection of Frequencies," against the acceptance criteria in SRP Section 3.7.3.II.4, which states that to avoid resonance, the fundamental frequencies of components and equipment should preferably be selected to be less than one-half or more than twice the dominant frequencies of the support structure, or otherwise the equipment is adequately designed for the applicable loads.

The staff reviewed PSAR Section 3.7.3.4, and noted the following:

- The cutoff frequency selected (follows PSAR Section 3.7.2.1.1) for the time history and response spectrum analyses ensures that all significant modes (excludes higher modes with less than 10 percent cumulative contribution to total response) are included in the superposition.
- For seismic loads, modes up to 100 Hz frequency are included.
- To avoid adverse resonance effects, where practical, components and equipment are designed/selected such that their fundamental frequencies are approximately less than half or more than twice the dominant frequency of the support structure. Moreover, the equipment is qualified by analysis, testing, or both such that it is adequately designed for applicable dynamic loads.

Based on the above review, the staff finds the applicant's basis for the selection of frequencies acceptable because it is consistent with the guidance in SRP Section 3.7.3.II.4.

3.7.3.3.5 Analysis Procedure for Damping

3.7.3.3.5.1 *RPV and Internals Damping*

The staff reviewed PSAR Section 3.7.3.5.1 regarding analysis procedure for damping for RPV and internals against the acceptance criteria in SRP Section 3.7.3.II.5, which refers to SRP Section 3.7.II.13.

From review of PSAR Section 3.7.3.5.1, "RPV and Internals Damping," the staff noted the following:

- PSAR Table 3.7-6, as amended by CPA Supplement 4, dated December 2, 2025 (TVA 2025-TN13047, TVA 2025-TN13045), lists the structural damping values used in the seismic analysis of the RPV and internals components using modal superposition method for steel construction, which are consistent with or conservative to that recommended in RG 1.61, Revision 2.

- The PSAR Section 3.7.3.5.1, as amended by CPA Supplement 4, dated December 2, 2025 (TVA 2025-TN13047, TVA 2025-TN13045), also states that fuel damping in Table 3.7-6 corresponds to bolted steel with bearing connection from RG 1.61 with additional conservatism added for consistency with previous BWR designs, including the economic simplified boiling water reactor (ESBWR) approved in NUREG-1966.
- Rayleigh damping is used for FE analysis of the RPV and internals when using direct integration method.

The staff finds that the damping values specified for RPV and internals components are consistent or conservative (smaller) to those specified in RG 1.61, Revision 2, for welded steel or bolted steel with bearing connections and, for fuel components, are consistent with those used previously in ESBWR design; thus, the values specified are acceptable for modal superposition analysis. The staff also finds the use of Rayleigh damping (proportional damping) with the direct integration method in the time domain is acceptable because it is a recommended approach in Section 3.5 of consensus standard ASCE 4-16 for structures composed of the same material or with similar damping characteristics, with the RPV and internals being primarily of the same material (i.e., steel). Therefore, the information provided in the PSAR for implementing damping for RPV and internals subsystems is adequate to meet the acceptance criteria in SRP Section 3.7.2.II.5.

3.7.3.3.5.2 *Damping for Piping, Equipment, and Equipment Support*

The staff reviewed PSAR Section 3.7.3.5.2 regarding analysis procedure for damping for piping, equipment, and equipment support against the acceptance criteria in SRP Section 3.7.3.II.5, which refers to SRP Section 3.7.II.13.

From review of PSAR Section 3.7.3.5.2, the staff noted that PSAR Table 3.7-7, as amended by CPA Supplement 4, dated December 2, 2025 (TVA 2025-TN13047, TVA 2025-TN13045), provides the damping coefficient values used in the seismic analysis of the seismic Category 1 piping, equipment, equipment supports, and intermediate structures, which are consistent with RG 1.61, Revision 2, for piping.

The staff also noted that the damping values provided in PSAR Section 3.7.3.5, Tables 3.7-6 and 3.7-,7 are used as constant material damping values in spectral analyses (SRP Section 3.7.2.II.13(B)) or the Rayleigh damping limits for transient dynamic analyses (SRP Section 3.7.2.II.13(C), direct integration in time domain).

The staff finds that the damping coefficient values specified in PSAR Table 3.7-7 for seismic Category 1 piping, equipment, and equipment supports are acceptable because they are consistent with those specified in RG 1.61, Revision 2, for piping systems.

The staff also finds the use of constant (frequency independent) material damping in response spectrum analysis method or Rayleigh damping (proportional damping) with the direct integration time domain method is acceptable because they are recommended approaches in Section 3.5 of consensus standard ASCE 4-16 for structures composed of the same material or with similar damping characteristics, with the piping, equipment, and equipment supports being primarily of the same material (i.e., steel). Therefore, the information provided in the PSAR for implementing damping for piping, equipment, and equipment support subsystems is adequate to meet the acceptance criteria in SRP Section 3.7.2.II.5.

3.7.3.3.6 Three Components of Design Ground Motion

In PSAR Section 3.7.3.6, “Three Components of Design Ground Motion,” the applicant indicated that seismic responses resulting from the analysis of subsystems in response to three components of earthquake motions are combined in the same manner as the seismic response resulting from the analysis of structural systems, described in PSAR Section 3.7.2.6. The staff finds this approach acceptable because it is consistent with SRP Acceptance Criterion 3.7.3.II.6, which refers back to SRP Acceptance Criterion 3.7.2.II.6. The staff evaluation in Section 3.7.2.3.6 of this report also applies to PSAR Section 3.7.3.6.

3.7.3.3.7 Combination of Modal Response

In PSAR Section 3.7.3.7, “Combination of Modal Responses,” the applicant stated that modal combination procedures for seismic subsystems are in accordance with SRP Section 3.7.2.II.7 and RG 1.92, Revision 3. The staff reviewed PSAR Section 3.7.3.7 against SRP Section 3.7.3.II.7, which states that the acceptance criteria provided in SRP Section 3.7.2.II.7 are applicable. The staff finds that the information provided for combining modal responses of seismic subsystems is acceptable because it incorporates by reference SRP Acceptance Criterion 3.7.3.II.7 and follows the guidance in RG 1.92, Revision 3.

3.7.3.3.8 Interaction of Non-seismic Category I Subsystems with Seismic Category I SSCs

In PSAR Section 3.7.3.8, the applicant stated that the methodology used for evaluation and design of non-seismic Category I subsystems that could interact with seismic Category I SSCs is consistent with SRP Section 3.7.3.II.8. The staff finds this acceptable because it is consistent with the guidance in SRP Acceptance Criterion 3.7.3.II.8 incorporated by reference.

3.7.3.3.9 Multiple-Supported Equipment and Components with Distinct Inputs

The staff reviewed PSAR Section 3.7.3.3.9, “Multiple-Supported Equipment and Components with Distinct Inputs,” regarding equipment and components supported at several points where the seismic motions at the support points may be quite different, against the acceptance criteria in SRP Section 3.7.3.II.9. From the review of the PSAR section, the staff noted the following:

- The approach for analyzing equipment and components supported at several points by a single or two structure(s) is in accordance with the guidance in SRP Section 3.7.3.II.9.
- Time history direct integration, time history modal superposition, and response spectrum modal superposition methods may be used in multi-support excitation analysis.
- The uniform support motion (USM) method or the independent support motion (ISM) method may be used.
- The analyses are performed such that the inertial (primary) and static (secondary) stresses due to differential displacements are separated and evaluated in accordance with the design basis code (ASME BPVC, Section III).

The staff finds the applicant’s method for analyzing multiple-supported equipment and components acceptable on the basis it is implemented consistent with the acceptance criteria in SRP Section 3.7.3.II.9.

3.7.3.3.10 Use of Equivalent Vertical Static Factors

The staff reviewed PSAR Section 3.7.3.10, “Use of Equivalent Vertical Static Factors,” against the acceptance criteria in SRP Section 3.7.3.II.10, which refers to the acceptance criteria in SRP Section 3.7.2.II.10 that states in part that use of equivalent static factors for vertical seismic loads is acceptable if the acceptance criteria in SRP Section 3.7.2.I.B are satisfied in the vertical direction.

In PSAR Section 3.7.3.10, the applicant stated that the equivalent vertical static factors are used for seismic analysis of subsystems when the requirements for the static coefficient method in PSAR Section 3.7.3.1.3 are satisfied in the vertical direction.

The staff finds the method described in PSAR Section 3.7.3.10 for use of vertical static factors to determine vertical seismic loads for subsystems acceptable because it satisfies the acceptance criteria in SRP Section 3.7.3.II.10 by satisfying the SRP Section 3.7.2.I.B acceptance criteria in the vertical direction.

3.7.3.3.11 Torsional Effect of Eccentric Masses

The staff reviewed PSAR Section 3.7.3.11, “Torsional Effects of Eccentric Masses,” against the acceptance criteria in SRP Section 3.7.3.II.10, which states that when the torsional effect of an eccentric mass is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model.

In PSAR Section 3.7.3.11, the applicant stated that torsional effects of eccentric masses are considered in the modeling of subsystems as discussed in PSAR Section 3.7.3.3. Based on review of PSAR Section 3.7.3.3, the staff noted that torsional effects of eccentric or asymmetric masses are addressed for piping and equipment subsystems by including the eccentric masses with offsets in the finite element or lumped mass stick analytical models.

The staff finds the above description to be acceptable because it is consistent with SRP Acceptance Criterion 3.7.3.II.11.

3.7.3.3.12 Seismic Category I Buried Piping, Conduits, and Tunnels

In PSAR Section 3.7.3.12, the applicant stated that the BWRX-300 design does not include seismic Category I buried pipes, conduits, or tunnels. Therefore, no staff evaluation is required.

3.7.3.3.13 Methods for Seismic Analysis of Seismic Category I Concrete Dams

In PSAR Section 3.7.3.13, the applicant stated that the design does not include nor require the presence of a dam. Therefore, no staff evaluation is required.

3.7.3.3.14 Methods for Seismic Analysis of Aboveground Tanks

In PSAR Section 3.7.3.14, the applicant stated that the BWRX-300 design does not include seismic Category I above-ground tanks. Therefore, no staff evaluation is required.

3.7.3.3.15 Effects of Differential Building Movements

In PSAR Section 3.7.3.15, the applicant stated that subsystems are anchored and restrained to floor and walls of buildings that may have differential movements (displacements) during a seismic event that induce forces and moments in the system. The staff noted that the applicant categorized the resulting stress from restraint of free-end displacement of the subsystem as self-limiting secondary stress, and when it exceeds yield, minor distortions or deformations relieve the condition that caused the stress to occur. The staff finds that considering the stress induced by differential end displacements of the subsystem as a secondary self-relieving stress is acceptable based on limited permanent deformations (limit state LS-C) allowed in the system by Section 6.2 of NRC-approved topical report NEDO-33914-A. The staff evaluation of the methodology used to obtain differential displacements used in the evaluation of subsystems is described in Section 3.7.2.8 of this report.

3.7.3.4 Conclusion

The staff has reviewed the available information provided in the PSAR, as amended, and, for the reasons given above, concludes that the applicant's plant design seismic system modeling and analysis procedures, including soil-structure interaction (SSI), are acceptable to meet the SRP 3.7.3 acceptance criteria and thereby meet the relevant requirements of 10 CFR Part 50, Appendix A, GDC 2, and 10 CFR Part 50, Appendix S, with respect to the analysis methodology to demonstrate capability of the seismic Category 1 structures to withstand the effects of design-basis earthquakes. The staff review finds that the seismic design analysis procedures reflect appropriate consideration of the most severe earthquake recorded for the site with an appropriate margin consistent with the previously established seismic design parameters, consideration of two levels of earthquakes (with OBE set at one-third SSE), appropriate combination of the effects of normal and accident conditions with the effects of the seismic natural phenomena, the importance of the safety functions to be performed, and the use of suitable dynamic analyses or suitable qualification tests to develop design demands to demonstrate that SSCs can withstand the seismic and other concurrent loads. The staff finds that the applicant's seismic analysis procedures and descriptions adequately addressed the following considerations: (1) seismic analysis methods; (2) natural frequencies and response loads, basis for selection of frequencies, number of earthquake cycles; (3) analytical modeling procedures; (4) SSIs; (5) in-structure response spectra development; (6) three components of design ground motion; (7) combinations of modal responses; (8) interaction of non-Category 1 with Category 1 structures; (9) effects of parameter variations on responses; (10) equivalent vertical static factors; (11) methods to account torsional effects; (12) analysis procedure accounting for damping differences; and (13) seismic overturning and sliding. This provides reasonable assurance that the seismic analysis procedures are adequate for use in the analysis and design of the CRN-1 seismic Category I SSCs to withstand design seismic loadings. Therefore, the NRC staff finds that the information provided by the applicant is sufficient for preliminary design, meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section, and adequately supports the issuance of a CP pursuant to 10 CFR 50.35. The staff will confirm that the final design conforms to these seismic analysis methods and procedures during the evaluation of the CRN-1 FSAR.

3.7.4 Seismic Instrumentation

3.7.4.1 Introduction

PSAR Section 3.7.4 describes the seismic instrumentation program (including the instrumentation type, locations, characteristics, and maintenance) such that the seismic response of the CRN-1 nuclear power plant features important to safety can be evaluated promptly after an earthquake. PSAR Section 3.7.4 also describes the applicant’s pre-earthquake planning, shutdown, and restart criteria.

3.7.4.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR Part 50 (TN249), Appendix S, requires that suitable instrumentation be provided to promptly evaluate the seismic response of nuclear power plant features important to safety after an earthquake; this appendix also requires shutdown of the nuclear power plant.
- 10 CFR Part 20 (TN283) requires licensees to make every reasonable effort to maintain radiation exposure as low as reasonably achievable (ALARA).

SRP Section 3.7.4, Revision 3, “Seismic Instrumentation,” issued July 2014, lists acceptance criteria for the seismic instrumentation program that are adequate to meet the relevant requirements above. In addition, the following NRC regulatory guides (RGs) contain relevant guidance, as referenced in SRP Section 3.7.4, for meeting the requirements of 10 CFR 50 Appendix S and 10 CFR 20:

- RG 1.12, Revision 3, *Nuclear Power Plant Instrumentation for Earthquakes*, issued October 2017, for guidance on the seismic instrumentation criteria, including instrumentation type, locations, characteristics, and maintenance (NRC 2017-TN13040)
- RG 1.166, Revision 1, *Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake*, issued February 2020, regarding pre-earthquake planning actions; actions necessary to determine the need to shut down a nuclear plant; and the short- and long-term processes, inspections, and tests to demonstrate that a nuclear power plant is safe for restarting after a shutdown in response to an earthquake (NRC 2020-TN13041)
- RG 8.8, Revision 3, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable*, issued June 1978, for guidance to support verification that the layout and design of the seismic instrumentation are consistent with the guidance provided in RG 8.8 for meeting the requirements of 10 CFR 20.1101(b), as it relates to providing engineering controls based upon sound radiation protection principles to achieve occupational doses that are ALARA (NRC 1978-TN12865).

3.7.4.3 Technical Evaluation

The staff reviewed PSAR Sections 3.7.4.2 through 3.7.4.3, as well as PSAR Section 3.7.4.5, which describe the CRN-1 seismic instrumentation program. The staff determined that the proposed instrumentation program (including instrument type and locations, a discussion regarding instrument operability, characteristics, installation, remote indication, and maintenance) is generally consistent with RG 1.12. However, although the applicant’s proposed number and general location of instrumentation are consistent with RG 1.12, the staff determined that more specificity will be needed at the OL stage regarding the location of all proposed instrumentation as well as the bases for these locations. These instrumentation location details are also necessary for the staff to assess consistency with RG 8.8 and 10 CFR 20.1101(b).

The staff also reviewed PSAR Section 3.7.4.4 and the last paragraph of PSAR Section 3.7.4.5, which describe TVA's proposed pre-earthquake planning, shutdown, and restart criteria. Although applicant indicated that its pre-earthquake planning and post-earthquake actions are in accordance with RG 1.166 (including clarifications) and ANSI/AANS-2.23, the applicant's discussion in PSAR Section 3.7.4.4 focuses solely on ANSI/ANS-2.23, thereby not addressing ANSI/ANS-2.23 clarifications C.1 through C.7 provided in RG 1.166. In addition, the applicant's discussion of ANSI/ANS-2.23 focuses on Sections 6, 7.1, 7.2, 8.1, 8.2, and 9. As such, it is not clear if other pertinent sections of ANSI/ANS-2.23 will be considered. The staff determined that more clarity regarding the proposed pre-earthquake planning, shutdown, and restart criteria (per RG 1.166) will be needed at the OL stage.

3.7.5 Conclusion

The staff has reviewed the available information provided in the PSAR, and, for the reasons given above, concludes that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35. The staff also concludes that additional details regarding the applicant's proposed seismic instrumentation program can reasonably be left to the OL stage of the review; the staff will complete its review at that time make a determination with respect to 10 CFR Part 50, Appendix S as well as 10 CFR Part 20 per RG 8.8.

3.8 Design of Seismic Category I Structures

PSAR Section 3.8 addresses the design of the BWRX-300 seismic Category I structures consisting of the reactor building (RB), the containment, and containment internal structures, otherwise referred to as the integrated RB structures.

3.8.1 Steel-Plate Composite Containment Vessel

3.8.1.1 Introduction

PSAR, Revision 1, Section 3.8.1, describes the BWRX-300 steel-plate composite containment vessel (SCCV) containment, which comprises an SCCV, a Class MC steel containment closure head, and other Class MC components. The configuration of the SCCV containment is shown in PSAR Figure 3.8-1 and is completely enclosed within the RB. PSAR Section 3.8.1 is focused on the design basis for materials, design, fabrication, construction, examination, and testing of the SCCV portion of the BWRX-300 containment using DP-SC modules, which is a first-of-a-kind novel design feature.

The applicant amended PSAR Section 3.8.1 by CPA Supplement 4 dated December 2, 2025 (TVA 2025-TN13044), CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029).

Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), “Preliminary safety analysis report,” sub-paragraph (2), requires applicants to provide a summary description and discussion of the facility, with special attention to unusual or novel design features (in this case, SCCV) and principal safety considerations.
- 10 CFR 50.34(a), “Preliminary safety analysis report,” sub-paragraph (3)(ii), requires applicants to address the design bases (in this case for the containment) and the relation of the design bases to the principal design criteria, applicable to containment, and sub-paragraph (3)(iii) requires applicant to provide information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin of safety.
- 10 CFR 50.34(a)(4) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures (in this case, SCCV) of the facility including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures and components (in this case, SCCV) provided for the prevention of accidents and the mitigation of the consequences of accidents.
- 10 CFR 50.55a and GDC 1 require that the SCCV containment be designed, fabricated, erected, and tested to quality standards, commensurate with the importance of the safety functions to be performed.
- GDC 2 requires that the SCCV containment withstand the most severe natural phenomena, such as winds, tornadoes, floods, and earthquakes, and the appropriate combination of all loads
- GDC 4 requires that the SCCV containment be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- GDC 16 requires that the SCCV containment act as a leak-tight membrane to prevent the uncontrolled release of radioactive effluents to the environment.
- GDC 50 requires that the SCCV containment be designed with sufficient margin of safety to accommodate appropriate design loads including accident pressure and temperature.
- GDC 51, “Fracture prevention of containment pressure boundary,” requires that reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.
- GDC 52, “Capability for containment leakage rate testing,” requires that the reactor containment and other related equipment shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.
- GDC 53, “Provisions for containment testing and inspection,” requires that the reactor containment shall be designed to permit (1) periodic inspection of all important areas including penetrations, (2) a surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations.
- 10 CFR 50.34(f), “Additional TMI-related requirements,” specifically §50.34(f)(3)(v)(A) and (B).

- 10 CFR 50.44, “Combustible gas control for nuclear power reactors,” provides the requirements for combustible gas control for reactors. Specifically, for future water-cooled reactor applicants and licensees, §50.44(c)(5), “Structural analysis,” requires that an applicant perform an analysis using an NRC-accepted analytical technique that demonstrates containment structural integrity under an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning.
- 10 CFR Part 50, Appendix B, “Quality assurance criteria for nuclear power plants,” requires every applicant for a construction permit, to include in its preliminary safety analysis report a description of the quality assurance (QA) program to be applied to the design, fabrication, construction, and testing of the safety-related SSCs of the facility.
- 10 CFR Part 50, Appendix J, “Primary reactor containment leakage testing for water-cooled power reactors,” requires that primary reactor containments meet the containment leakage test requirements set forth in this appendix.
- 10 CFR Part 50, Appendix S, “Earthquake engineering criteria for nuclear power plants.”

The following NRC SRP and regulatory guides (RGs) and other guidance documents also contain relevant guidance to satisfy requirements for this review:

- SRP Section 3.8.1, Revision 4 (September 2013), “Concrete Containment,” provides guidance for materials, loads and load combinations, design and analysis procedures, construction, examination, and testing related to concrete containments or to concrete portions of steel/concrete containments.
- RG 1.136, Revision 4 (NRC 2021-TN13098), *Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments*, describes an approach that is acceptable to the NRC staff to meet regulatory requirements for materials, design, construction, fabrication, examination, and testing of concrete (reinforced or prestressed) containments in nuclear power plants. GDC 1, 2, 4, 16, and 50 are applicable to NRC RG 1.136 (NRC 2021-TN13098).
- RG 1.243, Revision 0, *Safety-Related Steel Structures and Steel-Plate Composite (SC) Walls for Other Than Reactor Vessels and Containments*, describes a method acceptable to the NRC staff for compliance with regulations for the design, fabrication, and erection of safety-related steel structures and SC walls for other than reactor vessels and containments (NRC 2021-TN13112). 10 CFR Part 50, Appendix A, GDC 1, 2, and 4; 10 CFR Part 50, Appendix B; and 10 CFR Part 50, Appendix S, are applicable. This guide endorses, with exceptions and clarifications, the procedures and standards of the ANSI/AISC N690-18 code, including provisions for SC walls in Appendix N9.
- RG 1.216, Revision 0, *Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure*, August 2010 (NRC 2010-TN13106).
- RG 1.28, Revision 6, *Quality Assurance Program Criteria (Design and Construction)*, September 2023 (NRC 2023-TN13105).
- RG 1.54, Revision 3, *Service Level I, II, III and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants*, April 2017 (NRC 2010-TN8358).
- Licensing Topical Report (LTR) NEDC-33926P/NEDO-33926P-A, Revision 4, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building (RB) Structural Design” (proprietary/non-proprietary) (ADAMS Accession No. ML25351A085 package, ML25351A087 proprietary; GE Hitachi 2025-TN13024 public); NRC Safety

Evaluation is included as the front matter of the LTR NEDC-33926P/NEDO-33926P-A version.

- DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications," Interim Staff Guidance, October 2022 (NRC 2022-TN12894) provides clarifying and supplemental guidance to the SRP written for combined licenses when used for CP reviews.

3.8.1.2 Technical Evaluation

3.8.1.2.1 Steel-Plate Composite Containment Vessel (SCCV)

The staff reviewed PSAR Section 3.8.1, "Steel-Plate Composite Containment Vessel," as amended by CPA Supplements 4 and 5, against the identified regulatory requirements, the regulatory guidance in SRP Section 3.8.1, "Concrete Containment"; RG 1.136, Revision 4, as applicable to SCCV with no reinforcement or tendons; and RG 1.243, Revision 0, and compliance with NRC-approved design-specific LTR NEDC-33926P-A as an acceptable way to satisfy the requirements identified in the regulatory evaluation. However, the staff did not conduct for CP review a structural design audit indicated in SRP Section 3.8.1.II.4.L, described in Appendix B to SRP Section 3.8.4, because the structural design of the SCCV is incomplete and in progress during the review of the CPA.

The staff reviewed PSAR Section 3.8.1, as amended, and noted the CRN-1 BWRX-300 SCCV structure consists of a cylindrical wall, mat foundation, and top slab. The SCCV is constructed using steel-plate composite modules with diaphragm plates referred to as diaphragm plate steel-plate composite (DP-SC) modules, which are a novel first-of-a-kind design feature. The configuration, general arrangement, and approximate overall dimensions of the containment are shown in PSAR Figures 3.8-2, 3.8-8, 1.2.-2, and 1.2-3. PSAR Section 3.8.1 also describes the primary functions and materials used of the containment. Based on the above review, the NRC staff finds the PSAR provided an adequate summary description and discussion of the BWRX-300 containment structure, with special focus on novel design features (in this case, SCCV using DP-SC modules) and its principal safety functions, thereby satisfying the requirements in 10CFR50.34(a)(2) and (a)(3)(iii).

The staff reviewed PSAR Sections 3.8.1.1 through 3.8.1.7 as amended, which respectively provide description of the SCCV; applicable codes, standards, and specifications including jurisdictional boundary; loads and load combinations; design and analysis procedures, including corrosion prevention; structural acceptance criteria; materials, quality control, and special construction techniques; and testing and in-service surveillance requirements, including pre-service structural integrity test, preservice and in-service inspection and integrated leak-rate testing.

The staff noted the above PSAR sections for the SCCV reference and the information provided are based on the detailed methodology for design and construction of DP-SC modules in Section 6.0 and related sections of LTR NEDC-33926P-A, Revision 4, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building (RB) Structural Design." The staff reviewed and approved this LTR with limitations and conditions (L&Cs) in NRC Safety Evaluation available in ADAMS Accession Nos. ML25268A140 - Proprietary, NRC 2025-TN13032 – Public, and included in the front matter of the -A version, as an acceptable way of meeting the identified regulatory requirements (i.e., 10 CFR 50.34(f); 10 CFR 50.44; 10 CFR 50.55a; GDC 1, 2, 4, 16, 50, 51, 52, and 53; and 10 CFR 50, Appendices B, J and S) for SCCV containment using DP-SC modules. Based on the staff review of PSAR Sections 3.8.1.1

through 3.8.1.7, as amended, and the use of the design and construction methodology in NRC-approved LTR NEDC-33926P-A including associated limitations and conditions (L&Cs) evaluated below, the staff finds that the PSAR provides adequate information to satisfy the requirements in 10CFR50.34(a)(iii) for issuance of a CP. Additionally, based on the staff evaluation documented in Section 3F of this report of the preliminary analysis and evaluation results for the containment in PSAR Appendix 3F, the staff finds that the requirements of 10 CFR 50.34(a)(4) are also met.

In its safety evaluation of the LTR, the NRC staff concluded that "the proposed design-specific methodology/approach for materials, design, fabrication, construction, testing and examination of the BWRX-300 SCCV and RB using DP-SC modules, as described in the -A version of the LTR, are reasonable and adequate as an acceptable way of meeting applicable regulations identified in SE Section 2.0, subject to the limitations and conditions as provided in Section 8.0."

The staff notes that LTR addresses conformance with the regulatory guidance, which includes the SRP and RGs listed in Section 3.8.1.2, "Regulatory Evaluation" of this report. The methodology described in the LTR for design and construction of SCCV and RB using DP-SC modules was developed by adapting the provisions of existing industry codes ASME BPVC, Section III, Division 2 (endorsed in RG 1.136), and ANSI/AISC N690-18 (endorsed in RG 1.243), and supplemented by a National Reactor Innovation Center (NRIC) test program and cognizant published literature. ASME BPVC, Section XI, Section IWE and Section IWL are adapted for in-service inspection of the SCCV.

The staff notes that several of the L&Cs in Section 8.0 of the LTR are expected to be addressed by the applicant during the CP stage. These are addressed and referenced by the applicant in the PSAR updates in CPA Supplement 5 (NRC 2026-TN13028 – package, TVA 2026-TN13029) and are evaluated for compliance in Section 3.8.1.3.2 of this report below.

The staff notes that LTR NEDC-33926P-A conforms to the regulatory guidance in SRP Section 3.8.1 and RG 1.136 to the extent applicable to SCCV with no reinforcement and tendons and the guidance in RG 1.243, Revision 0. Therefore, the staff review of PSAR Section 3.8.1 is based on its review of the LTR. The staff finds that the information described in PSAR Section 3.8.1, as amended, for materials, design, fabrication, construction, testing, and examination of the BWRX-300 SCCV and RB using DP-SC modules is consistent with LTR NEDC-33926P-A. As evaluated in Section 3.8.1.3.2 of this report below, the applicant has adequately demonstrated satisfying the associated L&Cs 8.3, 8.4, 8.6, 8.7, and 8.11, which are among those required to be addressed in the CP stage. Further, for L&Cs 8.12 and 8.16 that also require compliance at the CP stage, there are additional pending actions for the applicant to demonstrate compliance, and therefore the staff imposed CP conditions L-CP-3.8-1 and L-CP-3.8-2 discussed in Section 3.8.1.3.2 of this report to be satisfied prior to start of nuclear construction using DP-SC modules. The staff thus finds that when CP conditions L-CP-3.8-1 and L-CP-3.8-2 are satisfied, Section 3.8.1 of the PSAR adequately meets the acceptance criteria in SRP Section 3.8.1.II and the level of detail is acceptable for issuance of a construction permit.

3.8.1.2.2 Staff Review of Compliance with Limitations and Conditions (L&Cs) Associated with LTR NEDC-33926P-A, Revision 4

PSAR Sections 1.9, 3.1, 3.5, 3.7, 3.8, 6.2, and 9.1 and Appendices 3B, 3F, 3G, and 3H reference NRC-approved LTR NEDC-33926P-A, Revision 4, "Steel-plate Composite (SC) Containment Vessel (SCCV) and RB Structural Design" (ML25351A087 proprietary; GE Hitachi

2025-TN13024 public), which is a novel design feature for CRN-1. Section 8.0 of the NRC SE, included as the front matter of LTR NEDC-33926P-A, specifies that an applicant referencing the LTR in a license application must demonstrate compliance with the 17 L&Cs stated therein or provide adequate justification for any deviations. Six of these L&Cs (8.3, 8.4, 8,6, 8,7, 8.12, and 8.16) are required to be addressed in the CPA stage, and the remaining may be addressed in a future licensing action (e.g., operating license application).

The PSAR, as amended by CPA Supplement 5 (TVA 2026-TN13029) and CPA Supplement 6 (TVA 2026-TN13031), addressed and referenced in applicable sections that L&Cs 8.3, 8.4, 8,6, 8,7, 8.12, and 8.16 are required to be complied with at the CPA stage. Additionally, the PSAR update also addressed and referenced L&C 8.11.

The staff reviewed the PSAR sections in CPA Supplement 5 and CPA Supplement 6 to verify compliance with each of the above stated L&Cs 8.3 (section thickness not to exceed 60 inches), 8.4 (specified compressive strength of concrete infill not to exceed 8 ksi), 8.6 **[[** (maximum surface temperature for use of LTR equation 5-19 not to exceed 482°F (250°C) **]]**, 8.7 (regarding several acceptance criteria for impact and impulsive loading), 8.11 (radius of curvature to wall panel thickness ratio of curved DP-SC walls to be greater than 2.0), 8.12 (selected corrosion protection), and 8.16 (ventholes). The staff evaluation of these L&Cs is documented below.

L&C 8.3 requires the DP-SC section thickness (t_{sc}) to not exceed 60 inches. The staff reviewed PSAR Figure 3.8-8; PSAR Sections 3F.1, 3G.1, and 3H.1; and PSAR Tables 3F-1, 3G-1, and 3H-1, as amended by CPA Supplement 5, and verified that the DP-SC member thicknesses for seismic Category 1 structures do not exceed 60 inches. The staff therefore finds that the applicant has demonstrated compliance with L&C 8.3.

L&C 8.4 requires the upper bound (maximum) specified concrete compressive strength (f’c) of concrete infill for DP-SC sections shall not exceed 8 ksi. The staff reviewed PSAR Sections 3F.1, 3G.1, and 3H.1, and PSAR Tables 3F-1, 3G-1, and 3H-1, as amended by CPA Supplement 5, and verified that the concrete compressive strength specified for concrete infill of DP-SC sections for seismic Category 1 structures do not exceed 8 ksi. The staff therefore finds that the applicant has demonstrated compliance with L&C 8.4.

L&C 8.6 requires that **[[**
]]. The staff reviewed PSAR Section 3E.1.12, “Thermal Conditions,” as amended by CPA Supplement 5, and verified that the maximum design internal temperature for accident conditions considered for the design of the containment is 330°F, which is **[[** **]]**. The staff therefore finds that the applicant has demonstrated compliance with L&C 8.6.

L&C 8.7 related to acceptance criteria for impact and impulsive loading states that

(a) in LTR Section 5.8.1.3, Table 5-2, the allowable ductility ratio for in-plane shear (shear walls) shall be limited to 1.5 (and not 3.0) consistent with Table 14 of referenced International Atomic Energy Agency (IAEA) SR No.87; (b) also, the statement in the second bullet of LTR Section 5.8.1.3 for DP-SC containment under DBAs is further clarified: for normal and severe environmental load categories, the allowable limits for ductility, support rotation, and strain shall not exceed those for superficial damage in LTR Tables 5-2 and 5-3, and, for abnormal, extreme environmental, and abnormal and extreme environmental load categories, the allowable limits for ductility, support rotation, and strain shall not exceed those for limited damage in LTR Tables

5-2 and 5-3; (c) additionally, for flexure controlled DP-SC components, in accordance with Footnote (2) to LTR Table 5-2, and regulatory position C 11.1.4 in RG 1.243, the criteria in terms of support rotations from Table 5-2, in terms of ductility from LTR Table 5-2 and in terms of strains from LTR Table 5-3, shall all be met to control damage.

The staff reviewed PSAR Section 3.5.3.4 and PSAR Tables 3.5-5a and 3.5-5b referenced therein, each as amended by CPA Supplement 5 (TVA 2026-TN13029), and noted that the new PSAR Tables 3.5-5a and 3.5-5b incorporated corresponding LTR NEDC-33926P-A Tables 5-2 and 5-3 and further explicitly included the modifications required by items (a), (b), and (c) of L&C 8.7 above. The L&C 8.7(a) ductility ratio criterion for in-plane shear for shear walls was specified as 1.5 (and not 3.0) in PSAR Table 3.5-5a. The L&C 8.7(b) criteria were captured in footnotes (10) and (11) in PSAR Table 3.5-5a and footnotes (2) and (3) in PSAR Table 3.5-5b. The L&C 8.7(c) criterion was captured in footnote (2) of PSAR Table 3.5-5a. Based on this review, the staff finds that criteria in PSAR Section 3.5.3.4 and Tables 3.5-5a and 3.5-5b, as amended, are compliant with the modifications required by L&C 8.7 and therefore acceptable.

L&C 8.11 requires that the curved DP-SC walls of the integrated RB (including SCCV and containment inerting system [CIS]) shall be designed and detailed to have a radius of curvature to wall panel thickness ratio greater than 2.0. It further requires that any residual stresses and strains resulting from rolling of the curved plates shall be evaluated and incorporated in detailed design.

The staff reviewed PSAR Section 3.8.4.1.4, as amended by CPA Supplement 5, with regard to L&C 8.11 and noted that the design criteria for integrated RB structures specified that curved walls are designed and detailed to have a radius-to-wall panel thickness greater than 2.0 and that any residual stresses and strains resulting from rolling of the curved plates are evaluated. The staff also reviewed PSAR Section 3F.3, as amended by CPA Supplement 5, and noted that an evaluation was performed to quantify the effects of residual stresses due to rolling on the curved plates within the IRB structures. The staff further noted that the evaluation determined that residual stresses and strains resulting from rolling of the curved plates are minimal and do not need to be accounted for further in the design of curved walls. During the audit, the staff audited GE Hitachi document DBR-0087606, Revision 2, “Effect of Residual Stresses Due to Rolling for the Curved Steel Sections for RB and SCCV on the Overall Stress State, BWRX-300 Reactor Building (U71) and SCCV (T10)”; verified that it quantified the effects of residual stresses due to rolling of the curved faceplates; and concluded the effects of residual stress pattern through faceplate thickness will be minimal on the ability of faceplates to reach membrane yield strength in tension and compression. The staff also confirmed during the audit that the smallest radius-to-thickness ratio, which is for the RPV pedestal, was above 2.0. Based on the above review, the staff finds that the applicant has adequately addressed compliance with L&C 8.11.

L&C 8.12 requires that the applicant referencing LTR NEDC-33926P-A in a CPA shall specify details of and justify adequacy of the selected combination of corrosion protection measure(s) from LTR Section 5.15 that will be implemented for the plant for corrosion protection of steel-plate composite (SC)/DP-SC modules.

The staff reviewed PSAR Sections 3.8.1.4.8, 3.8.4.1.1, and 3.8.4.1.6, as amended by CPA Supplement 5 and CPA Supplement 6 (TVA 2026-TN13031), and noted that while the PSAR indicates that, among other methods for different components (e.g., pool liners), a sacrificial thickness (not considered for strength or stiffness) is added to the inner and outer faceplates as corrosion tolerance to the below grade portion of the RB exterior wall and to the common mat foundation to compensate for potential corrosion, the corrosion protection plan for the DP-SC modules is yet to be finalized. The PSAR states that once the corrosion protection plan to be implemented for DP-SC modules is finalized, justification for the adequacy of the selected combination of measures will be provided prior to commencing nuclear construction using DP-SC modules per L&C 8.12 of NEDC-33926P-A. Since this is a work in progress, the staff will track compliance with L&C 8.12 as CP Condition L-CP-3.8-1 and will defer its determination of adequate compliance with L&C 8.12 when the corrosion protection plan to be implemented for DP-SC modules is finalized and justification for the adequacy of selected measures is provided by the applicant for staff review at a later date prior to start of nuclear construction using DP-SC modules. This is captured as CP Condition L-CP-3.8-1 below:

CP Condition L-CP-3.8-1: TVA shall demonstrate compliance with L&C 8.12 (related to identification and justification of adequacy of the selected corrosion protection measure(s) for DP-SC modules) in the NRC staff's safety evaluation included in the front matter of LTR NEDC-33926P-A, Revision 4 (ML25351A087 proprietary; GE Hitachi 2025-TN13024 public). TVA shall submit for completion of NRC staff review the finalized corrosion protection plan to be implemented for DP-SC modules of the integrated RB structures with justification for the adequacy of the selected combination of measures prior to commencing nuclear construction using DP-SC modules per L&C 8.12 of NEDC-33926P-A.

L&C 8.16 requires the detailed design implementation of vent holes when used should be made available for NRC staff review as part of a CPA referencing the LTR. The staff reviewed PSAR Section 3.8.1.4.1, as amended by CPA Supplement 4 dated December 2, 2025 (TVA 2025-TN13044) and CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029), and noted it stated that vent holes are used in the integrated RB steel-plate composite structures and the design of ventholes complies with the requirements of Sections 5.14 and 6.21 of NEDC-33916P-A, which the staff finds acceptable to partially address L&C 8.16 because ventholes will be provided and designed to NRC-approved LTR provisions. However, the PSAR, as amended, also states that detailed design implementation of vent holes (which per L&C 8.16 should be provided for NRC review during CP stage) is still progressing for CRN-1 and that L&C 8.16 of NEDC-33926P-A will be addressed prior to commencing nuclear construction using DP-SC modules. Since this is a work in progress, the staff will track compliance with L&C 8.16 as CP Condition L-CP-3.8-2 and will defer its determination of adequate compliance with L&C 8.16 when the detailed design implementation of vent holes is provided for staff review at a later date prior to start of nuclear construction using DP-SC modules. This is captured as CP Condition L-CP-3.8-2 below:

CP Condition L-CP-3.8-2: TVA shall demonstrate compliance with L&C 8.16 (related to design implementation of vent holes) in the NRC staff's safety evaluation included in the front matter of LTR NEDC-33926P-A, Revision 4 (ML25351A087 proprietary; GE Hitachi 2025-TN13024 public), and submit for completion of NRC staff review the completed detailed design implementation of ventholes for the integrated RB structures of DP-SC construction prior to start of nuclear construction using DP-SC modules per L&C 8.16 of NEDC-33926P-A.

Based on the above review, the staff concludes that the applicant has adequately demonstrated compliance with L&Cs 8.3, 8.4, 8.6, 8.7, and 8.11, which, with exception of 8.11, are required to be addressed in the CPA stage. Additionally, L&Cs 8.12 and 8.16, also required to be addressed at the CP stage, are subject to CP Conditions L-CP-3.8-1 and L-CP-3.8-2, and the staff will defer its determination of adequate compliance with these L&Cs following staff review when additional information related to these L&Cs are provided for staff review prior to start of nuclear construction using DP-SC modules.

The staff will review TVA compliance with all other L&Cs on LTR NEDC-33926P-A during FSAR review (i.e., operating license application stage).

3.8.1.3 Conclusion

The staff has reviewed the available information provided in the PSAR Sections 3.8.1 and Appendix 3F, as amended, and, for the reasons given above and subject to CP Conditions L-CP-3.8-1 and L-CP-3.8-2, concludes that the design bases of SCCV containment are acceptable to meet the relevant requirements of 10 CFR 50.34(f); 10 CFR 50.44; 10 CFR 50.55a; GDC 1, 2, 4, 16, 50, 51, 52, and 53; and 10 CFR 50, Appendix B using RGs 1.136, 1.243; NRC-approved LTR NEDC-33926P-A and RG 1.216 referenced therein; and industry standards ASME BPVC, Section III, Division 2, and ANSI/AISC N690-18, to provide reasonable assurance that the SCCV will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. The applicant has provided information related to the materials of construction, general arrangement, and approximate dimensions of the SCCV, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety. Therefore, based on conformance with regulatory guidance identified above and NRC-approved LTR NEDC-33926P-A, the staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(2), (a)(3)(ii), (a)(3)(iii) and (a)(4) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35. The staff will confirm that the final SCCV design conforms to the design bases during the evaluation of the CRN-1 FSAR.

The staff also concludes that the applicant has adequately demonstrated compliance with L&Cs 8.3, 8.4, 8.6, 8.7, and 8.11 on LTR NEDC-33926P-A, which, with exception of L&C 8.11, are required to be addressed in the CPA stage. Additionally, L&Cs 8.12 and 8.16, also required to be addressed at the CP stage, are subject to CP Conditions L-CP-3.8-1 and L-CP-3.8-2, stated previously.

3.8.2 Class MC Steel Components of the BWRX-300 Containment

3.8.2.1 Introduction

PSAR Section 3.8.2 describes the design basis for the Class MC steel components of the BWRX-300 containment comprising the containment closure head, two containment airlocks, and the containment penetrations.

The applicant amended PSAR Section 3.8.2 by CPA Supplement 4, dated December 2, 2025 (TVA 2025-TN13044), and CPA Supplement 5, dated January 7, 2026 (TVA 2026-TN13029).

3.8.2.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), “Preliminary safety analysis report,” sub-paragraph (3)(ii), requires applicants to address the design bases (in this case, for the Class MC components of containment) and the relation of the design bases to the principal design criteria, applicable to containment, and sub-paragraph (3)(iii) requires applicants to provide information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin of safety.

- 10 CFR 50.55a, "Codes and Standards," paragraph (a)(1) incorporates by reference (IBR) the ASME BPVC, Section III, Division 1, standard up to the 2021 edition, but is limited by those provisions identified in paragraph (b)(1), *Conditions on ASME BPV Code Section III,* and specifically the Section III conditions (e.g., (b)(1)(x), (xii)), which apply to the provisions of the code including Subsection NE of the 2021 edition.
- GDC 1 requires that the Class MC portions of containment be designed, fabricated, erected, and tested to quality standards, commensurate with the importance of the safety functions to be performed.
- GDC 2 requires that the Class MC portions of containment withstand the most severe natural phenomena, such as winds, tornadoes, floods, and earthquakes, and the appropriate combination of all loads
- GDC 4 requires that the Class MC portions of containment are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- GDC 16 requires that the Class MC portions of containment act as a leak-tight membrane to prevent the uncontrolled release of radioactive effluents to the environment.
- GDC 50 requires that the Class MC portions of containment be designed with sufficient margin of safety to accommodate appropriate design loads including accident pressure and temperature.
- GDC 51, "Fracture prevention of containment pressure boundary," requires that reactor containment boundary shall be designed with sufficient margin to assure that, under operating, maintenance, testing, and postulated accident conditions, (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.
- GDC 52, "Capability for containment leakage rate testing," requires that the reactor containment and other related equipment shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.
- GDC 53, "Provisions for containment testing and inspection," requires that the reactor containment shall be designed to permit (1) periodic inspection of all important areas including penetrations, (2) a surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations.
- 10 CFR 50.34(f), "Additional TMI-related requirements," specifically §50.34(f)(3)(v)(A) and (B).
- 10 CFR 50.44, "Combustible gas control for nuclear power reactors," provides the requirements for combustible gas control for reactors. Specifically, for future water-cooled reactor applicants and licensees, §50.44(c)(5), "Structural analysis," requires that an applicant perform an analysis using an NRC-accepted analytical technique that demonstrates containment structural integrity under an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning.
- 10 CFR Part 50, Appendix B, "Quality assurance criteria for nuclear power plants," requires every applicant for a construction permit to include in its preliminary safety analysis report a description of the QA program to be applied to the design, fabrication, construction, and testing of the safety-related SSCs of the facility.

- 10 CFR Part 50, Appendix J, "Primary reactor containment leakage testing for water-cooled power reactors," requires that primary reactor containments meet the containment leakage test requirements set forth in this appendix.

The following NRC SRP and regulatory guides (RGs) and other guidance documents and standards also contain relevant guidance to satisfy requirements for this review:

- SRP Section 3.8.2, Revision 3 (May 2010), "Steel Containment," provides guidance for materials, loads and load combinations, design and analysis procedures, construction, examination, and testing related to steel containments or to other Class MC steel portions of steel/concrete containments.
- RG 1.57, Revision 2, *Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components*, May 2013, describes an approach that the NRC staff considers acceptable for use in designing metal primary reactor containment system components to demonstrate structural integrity (NRC 2013-TN13111). GDC 1, 2, 4, 16, and 50 are applicable to NRC RG 1.57.
- RG 1.216, Revision 0, *Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure*, August 2010 (NRC 2010-TN13106).
- RG 1.28, Revision 6, *Quality Assurance Program Criteria (Design and Construction)*, September 2023 (NRC 2023-TN13105).
- RG 1.54, Revision 3, *Service Level I, II, III and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants*, April 2017 (NRC 2010-TN8358).
- RG 1.84, *Design, Fabrication, and Materials Code Case Acceptability, ASME Section III*, USNRC, Washington, DC (NRC 2024-TN13087).
- DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications," Interim Staff Guidance, October 2022 (NRC 2022-TN12894).
- ASME BPVC, Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NE, "Class MC Components," 2021 edition.
- ASME BPVC, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, Subsection IWE.

3.8.2.3 Technical Evaluation

The staff reviewed PSAR Section 3.8.2, "Class MC Steel Components of the BWRX-300 Containment," as amended by CPA Supplements 4 and 5, against the identified regulatory requirements; the regulatory guidance in SRP Section 3.8.2, Revision 3, "Steel Containment"; and RG 1.57, Revision 2, and industry standard ASME BPVC, Section III, Division 1, Subsection NE endorsed in 10 CFR 50.55a, as an acceptable way to satisfy the regulatory requirements identified in the regulatory evaluation. However, the staff did not conduct for CP review a structural design audit indicated in SRP Section 3.8.2.II.4.E described in Appendix B to SRP Section 3.8.4 because the structural design of the Class MC containment components is incomplete and in progress during the review of the CPA.

The staff reviewed PSAR Section 3.8.2, as amended, and noted the Class MC components of the BWRX-300 containment include a closure head at the top, two personnel airlocks, and penetrations all of steel material. The configuration, general arrangement and code jurisdictional boundary of these components are illustrated in PSAR Figures 3.8-1 through 3.8-5.

The staff reviewed PSAR Sections 3.8.2.1 through 3.8.2.7, as amended, and noted they provide descriptions of the design basis and criteria for Class MC components of containment; applicable codes, standards, and specifications including jurisdictional boundary; loads and load combinations; design and analysis procedures; structural acceptance criteria; materials and quality control, with no special construction techniques; testing and in-service surveillance requirements, including pre-service structural integrity test and pre-service and in-service inspection. The PSAR states that the containment closure head, access openings, and penetrations of the containment are analyzed, designed, fabricated, constructed, examined, and tested following the requirements of the ASME BPVC, Section III, Division 1, Subsection NE (2021 edition) endorsed in 10 CFR 50.55a, and conform to the regulatory guidance in SRP Section 3.8.2 and RG 1.57. The PSAR also indicates that in-service inspection of the Class MC components of containment will follow the requirements of ASME BPVC, Section XI, Subsection IWE. PSAR Figures 3.8-6 and 3.8-7 provide flowcharts of the design procedures for the containment closure head and other Class MC components, respectively. The staff notes that by complying with the material specifications and testing requirements in Article NE-2000 of ASME BPVC, Section III, Division 1, Subsection NE, endorsed in 10 CFR 50.55a, assure that the requirements of GDC 51 will be met for containment Class MC components. Additionally, complying with provisions for design, inspection and testing in accordance with ASME BPVC, Section complying with the material specifications and testing requirements in Article NE-2000 of ASME BPVC, Section III, Division 1, Subsection NE, and an in-service inspection program in accordance with ASME BPVC, Section XI, Subsection IWE required by 10 CFR 50.55a assure that the requirements of GDC 52 and GDC 53 will be met.

The staff notes that the 2021 edition of the ASME BPVC, Section III, Division 1, Subsection NE (the proposed code of record) is incorporated by reference (IBR) with conditions in 10 CFR 50.55a(a) with additional guidance provided in RG 1.57. The staff review of PSAR Section 3.8.2 and Table 1.9-20, as amended by CPA Supplement 5, dated January 7, 2026 (NRC 2026-TN13028 (package); TVA 2026-TN13029 (PSAR update), confirmed that the descriptions for containment Class MC components in the PSAR subsections identified above were consistent with the requirements ASME BPVC, Section III, Division 1, Subsection NE (2021 edition) and the guidance in RG 1.57. In its response to audit question A-3.8-4 (TVA 2026-TN13051), the applicant clarified that the phrase “Class MC components backed by concrete” used in PSAR Section 3.8.2 refers to items such as sleeves for penetrations going through the SCCV boundary with concrete on one side, as illustrated in PSAR Figure 3.8-5, which are designed in accordance with LTR NEDC-33926P-A. The staff finds this clarification acceptable because it is consistent with Sections 6.12 and 6.13 of NRC-approved LTR NEDC-33926P-A as follows. The staff noted that LTR Section 6.12, “Design and Detailing of Penetrations and Openings,” states that the design and detailing of penetrations are coordinated and meet requirements of CC-3740, “Penetration Assemblies,” of ASME BPVC, Section III, Division 2, and clarifies design and detailing of portions backed by concrete and those not backed by concrete (e.g., nozzle). The staff also noted that typical weld joint locations and joint categories are addressed in LTR Section 6.13, and in general penetrations are covered by Joint Categories C and D.

The staff confirmed that the PSAR indicates compliance with the applicable regulatory conditions in 10 CFR 50.55a(b)(1) (e.g., subparagraphs (b)(1)(x) and (b)(1)(xii)) associated with the use of the 2021 edition of the ASME BPVC, Section III-1, Subsection NE, incorporated by reference (IBR) in 10 CFR 50.55a(a)(1) based on statement in PSAR Section 3.8.2.2, as amended by Supplement 4, dated December 2, 2025 (TVA 2025-TN13045).

Based on the above review, the staff finds that the design basis and design criteria information described in PSAR Section 3.8.2, as amended, for materials, design, fabrication, construction, testing, and examination of the Class MC components (closure head, airlocks, and penetrations (with exception of penetration subcomponents backed by concrete) of the BWRX-300 are acceptable because they are consistent with the 2021 edition of the ASME BPVC, Section III, Division1, Subsection NE, including the applicable conditions in 10 CFR 50.55a(b)(1) and RG 1.57**.** Portions of penetrations (e.g., sleeves) backed by concrete are designed in accordance with NRC-approved LTR NEDC-33926P-A. Therefore, the staff finds that PSAR Section 3.8.2, as amended, adequately meets the acceptance criteria in SRP Section 3.8.2 and thereby meeting the requirements of 10 CFR 50.34(a)(3)(ii) and (a)(3)(iii) for issuance of a construction permit.

3.8.2.4 Conclusion

The staff has reviewed the available information provided in the PSAR Section 3.8.2, as amended and, for the reasons given above, concludes that the design bases and design criteria of the containment Class MC components (closure head, airlocks, and penetrations) are acceptable to meet the relevant requirements of 10 CFR 50.34(f); 10 CFR 50.44; 10 CFR 50.55a; GDC 1, 2, 4, 16, 50, 51, 52, and 53; and 10 CFR 50, Appendix B through the use of SRP 3.8.2,RG 1.57, RGs 1.28, and 1.216 as referenced within RG 1.57, and industry standards ASME BPVC, Section III, Division 1, Subsection NE (2021 edition) endorsed in 10 CFR 50.55a, to provide reasonable assurance that the containment Class MC components will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. The applicant has provided information related to the materials of construction and general arrangement of the containment Class MC components, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and (a)(3)(iii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35. The staff will confirm that the final design of Class MC components conform to the design bases during the evaluation of the CRN-1 FSAR.

3.8.3 Containment Internal Structures

3.8.3.1 Introduction

PSAR Section 3.8.3 describes the general configuration and design basis for the CRN-1 BWRX-300 containment internal structures (CIS), which are located inside the SCCV and include: (1) the RPV pedestal constructed of DP-SC modules and (2) the containment equipment and piping support structure (CEPSS) and two support platforms constructed of structural steel.

The applicant amended PSAR Section 3.8.3 by CPA Supplement 5, dated January 7, 2026 (TVA 2026-TN13029).

3.8.3.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), “Preliminary safety analysis report,” sub-paragraph (2), requires applicants to provide a summary description and discussion of the facility, with special attention to unusual or novel design features (in this case steel-plate composite RPV pedestal) and principal safety considerations.

- 10 CFR 50.34(a), “Preliminary safety analysis report,” sub-paragraph (3)(ii), requires applicants to address the design bases (in this case, for the CIS) and the relation of the design bases to the principal design criteria, applicable to CIS, and sub-paragraph (3)(iii) requires applicants to provide information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin of safety.
- 10 CFR 50.34(a)(4) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures (in this case, for CP focused on RPV pedestal) and transient conditions anticipated during the life of the facility, and the adequacy of structures and components (in this case, RPV pedestal) provided for the prevention of accidents and the mitigation of the consequences of accidents.
- 10 CFR 50.55a and GDC 1 require that the containment internal structures be designed, fabricated, erected, and tested to quality standards, commensurate with the importance of the safety functions to be performed.
- GDC 2 requires that the containment internal structures are capable of withstanding the most severe natural phenomena, such as winds, tornadoes, floods, and earthquakes, and the appropriate combination of all loads without loss of capability to perform their safety functions.
- GDC 4 requires that the containment internal structures be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- GDC 50 requires that the SCCV containment be designed with sufficient margin of safety to accommodate appropriate design loads.
- 10 CFR Part 50, Appendix B, “Quality assurance criteria for nuclear power plants,” requires every applicant for a construction permit, to include in its preliminary safety analysis report, a description of the QA program to be applied to the design, fabrication, construction, and testing of the safety-related SSCs of the facility.
- 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” paragraph (a)(1), requires licensees of a nuclear power plant to monitor the performance or condition of SSCs important to safety, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these SSCs remain capable of performing their intended functions.

The following NRC SRP and regulatory guides (RGs) and other guidance documents also contain relevant guidance to satisfy the requirements for this review:

- SRP Section 3.8.3, Revision 4 (September 2013), “Concrete and Steel Containment Internal Structures of Steel or Concrete Containments,” provides guidance for materials, loads and load combinations, design and analysis procedures, construction, examination, and testing related to containment internal structures of steel or concrete containments.
- RG 1.243, Revision 0, *Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments*, describes a method acceptable to the NRC staff for compliance with regulations for the design, fabrication, and erection of safety-related steel structures and steel-plate composite walls for other than reactor vessels and containments (NRC 2021-TN13112). 10 CFR Part 50, Appendix A, GDC 1, 2, and 4, and 10 CFR Part 50, Appendix B, are applicable. This guide endorses, with exceptions and

clarifications, the procedures and standards of the ANSI/AISC N690-18 code, including provisions for steel-plate composite walls in Appendix N9.

- RG 1.28, Revision 6, *Quality Assurance Program Criteria (Design and Construction)*, September 2023 (NRC 2023-TN13105).
- RG 1.54, Revision 3, *Service Level I, II, III and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants*, April 2017 (NRC 2010-TN8358). [Referenced in LTR NEDC-33926P-A]
- RG 1.160, Revision 4, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* (NRC 2018-TN7799).
- RG 1.70, Revision 3, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants – LWR Edition*, November 1978 (NRC 1978-TN12879).
- DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications," Interim Staff Guidance, October 2022 (NRC 2022-TN12894) provides clarifying and supplemental guidance to the SRP when used for CP reviews.
- Licensing Topical Report (LTR) NEDC-33926P/NEDO-33926P-A, Revision 4, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building (RB) Structural Design" (proprietary/non-proprietary) (ADAMS Accession No. ML25351A085 package, ML25351A087 proprietary; GE Hitachi 2025-TN13024 public); NRC Safety Evaluation is included as the front matter of the LTR NEDC-33926P/NEDO-33926P-A version; NRC Safety Evaluation of the LTR is also available at ADAMS Accession No. ML25268A140 – Proprietary, NRC 2025-TN13032 – Public.

3.8.3.3 Technical Evaluation

3.8.3.3.1 Description of Internal Structures

The staff reviewed the descriptive information of the CRN-1 CIS against the acceptance criteria in SRP Section 3.8.3.1.II.1, which refers to corresponding criteria in RG 1.70, Section 3.8.3.1. The staff noted that the general arrangement of the CIS (RPV, CEPSS, and support platforms) is illustrated in PSAR Figures 1.2-3, 3.8-1, and 3.8-8, and code jurisdictional boundary is shown in PSAR Figure 3.8-2.

3.8.3.3.1.1 *Reactor Pressure Vessel Pedestal*

The staff reviewed PSAR Section 3.8.3.1.1 and noted it states the following:

- The RPV pedestal is a cylindrical-shaped structure that supports the RPV and provides radiation shielding, constructed using DP-SC modules, and is integrated with the common mat foundation.
- The RPV pedestal is equipped with structural steel at the top where the RPV skirt is anchored using anchor bolts. It also provides structural support for the RPV bottom stabilizers, the control rod drive (CRD) housing outer springs, and the CEPSS.
- Openings are provided in the pedestal to permit the routing of piping to the RPV, to permit ISI of the RPV and piping, and to ensure personnel access into the under-vessel region.

The staff finds that the level of detail in the PSAR description of general arrangement, configuration, function, and principal features of the RPV pedestal structure is adequate for a CP because it is consistent with the criteria in SRP Section 3.8.3.II.1.

3.8.3.3.1.2 *Containment Equipment and Piping Support Structure*

The staff reviewed PSAR Section 3.8.3.1.2 and noted it states the following:

- The structural steel CEPSS is supported and connected vertically by the RPV pedestal and is laterally supported through tangential restraints to the SCCV as shown in PSAR Figure 3.8-1.
- Shear lug support connections with the SCCV are used to allow free thermal expansion of the CEPSS while resisting CEPSS lateral forces.
- The CEPSS consists of various structural components, such as beams and columns, and the framing includes vertical bracing members around the CEPSS outer perimeter (near SCCV wall) and inner perimeter (above the pedestal), and horizontal bracing members to transfer lateral forces to the SCCV wall.
- The CEPSS structure supports the RPV top stabilizers that provide horizontal bracing for the RPV. In addition, the CEPSS also supports piping, RIVs including actuators, and miscellaneous platforms needed for personnel access and equipment inspection and maintenance.

The staff finds that the level of detail in the PSAR description of general arrangement, configuration, function, and principal features of the CEPSS is adequate for a CP because it is consistent with the criteria in SRP Section 3.8.3.II.1.

3.8.3.3.1.3 *Support Platforms*

The staff reviewed PSAR Section 3.8.3.1.3 and noted it states the following:

- The steel support platforms are of structural steel and shown in PSAR Figure 3.8-1 and provide support for miscellaneous components, including fans and coolers associated with the containment cooling system, and for apparatus needed for equipment inspection and maintenance.
- The steel support platform beams are supported by the RPV pedestal and SCCV wall.

The staff finds that the level of detail in the PSAR description of general arrangement, configuration, and function of the CEPSS is adequate for a CP because it is consistent with the criteria in SRP Section 3.8.3.II.1.

Based on the above review in Section 3.8.3.3.1 of this report, the NRC staff finds the PSAR provided an adequate summary description and discussion of the BWRX-300 containment internal structure, with special focus on novel design features (in this case, the RPV pedestal using DP-SC modules) and its principal safety functions, and in Section 3.8.3.3.6 of this report materials of construction, thereby satisfying the requirements in 10 CFR 50.34(a)(2) and (a)(3)(iii).

3.8.3.3.2 Applicable Codes, Standards, and Specifications

The staff reviewed PSAR Section 3.8.3.2, as amended by CPA Supplement 5, on applicable codes and standards for materials, design, fabrication, erection/construction, examination, testing, and in-service inspection of the containment internal structures, against the acceptance criteria in SRP Section 3.8.3.1.II.2. The staff noted the following codes and standards are used:

- For the DP-SC RPV pedestal, Section 5 of LTR NEDC-33926P-A, which adapted and modified the provisions for steel-plate composite walls of ANSI/AISC N690-18, Appendix N9, as endorsed and modified by RG 1.243, Revision 0.
- For structural steel containment internal structures, ANSI/AISC N690-18, as endorsed in RG 1.243 (also see PSAR Section 3.8.3.5)

The staff finds that the applicant has identified the appropriate code and standard (ANSI/AISC N690-18) for the structural steel part of containment internal structures because it is endorsed in RG 1.243, Revision 3, and therefore is consistent with the acceptance criteria in SRP Section 3.8.3.1.II.2.

The staff also finds that the applicant has identified appropriate LTR NEDC-33926P-A and associated standards and guidance (N690-18, RG 1.243) for the DP-SC RPV pedestal because LTR NEDC-33926P-A is NRC-approved with limitations and conditions (L&Cs) in Section 8.0; NRC Safety Evaluation is available at ADAMS Accession Nos. ML25268A140 - Proprietary, NRC 2025-TN13032 – Public, but subject to CP Conditions L-CP-3.8-1 and L-CP-3.8-2 discussed in Section 3.8.1.3.2 of this report.

3.8.3.3.3 Loads and Loading Combinations

The staff reviewed PSAR Section 3.8.3.3 against the acceptance criteria in SRP Section 3.8.3.II.3.B applicable for steel and steel-plate composite structures and RG 1.243.

3.8.3.3.3.1 *Design Loads*

The staff reviewed PSAR Section 3.8.3.3.1 on design loads for containment internal structures and noted the following:

- The design loads also include the reactions from the RPV at the support locations on the containment internal structures and other bracket and attachment loads applicable during different plant conditions.
- Since RPV is included in the one-step FE model of the IRB, dead load and seismic load reactions from the RPV are obtained directly from the static and IRB seismic analyses.
- The design loads for the containment internal structures are shown in PSAR Table 3.8-7.

The staff finds the loads for design of the containment internal structures shown in PSAR Table 3.8-7 acceptable because they are based on design code ANSI/AISC N690-18 endorsed in RG 1.243, and they include the reaction loads at interfaces of the containment internal structures with the SCCV; therefore, they meet the SRP Section 3.8.3.II.3.B acceptance criteria.

3.8.3.3.3.2 *Load Combinations*

The staff reviewed PSAR Section 3.8.3.2.2 and noted that the design load combinations and load factors for the DP-SC RPV pedestal and CIS structural steel that form the containment internal structures are in accordance with ANSI/AISC N690, including the supplemental guidance of RG 1.243, and covered by the load combinations listed in PSAR Table 3.8-7.

The staff finds that the design loads and load combinations meet the acceptance criteria in SRP Section 3.8.3.II.3.B because they are consistent with design code AISC/N690-18 for steel structures, as supplemented by the regulatory positions in RG 1.243, which also applies to steel-plate composite/DP-SC structures.

3.8.3.3.4 Design and Analysis Procedures

The staff reviewed the design and analysis procedures for containment internal structures in PSAR Section 3.8.3.4 against the acceptance criteria in SRP Section 3.8.3.II.4. However, the staff did not conduct for CP review a structural design audit indicated in SRP Section 3.8.3.II.4.F described in Appendix B to SRP Section 3.8.4 because the structural design of the containment internal structures is incomplete and in progress during the review of the CPA.

From review of the design and analysis procedures in PSAR Sections 3.8.3.4.1 for RPV pedestal, 3.8.3.4.2 for CEPSS, and 3.8.3.4.3, the staff noted the following:

- The RPV pedestal, CEPSS, and support platforms of the containment internal structures are part of the IRB FE model and analyzed by the one-step analysis procedure in PSAR Section 3.8.4.1.4.
- RPV pedestal in the IRB model includes major openings and penetrations in accordance with ANSI/AISC N690-18, Appendix N9. The states of stress around these openings are determined by analytical techniques.
- The connections between the RPV pedestal, the RPV, the CEPSS, and support platforms are appropriately modeled to reflect the load transfer for vertical, thermal, and reaction forces and lateral loads to the mat foundation.
- The CEPSS supports and transfers vertical and lateral loads coming from the containment equipment and piping to the RPV pedestal and SCCV wall. The connections between the CEPSS, the RPV pedestal, and SCCV are appropriately modeled in the integrated RB FE model to reflect the load transfer for vertical and lateral loads.
- The support platforms’ vertical and lateral loads are supported by the RPV pedestal and SCCV wall and included in the IRB model.
- Preliminary design demands for critical sections of the containment internal structures are provided in Appendix 3G (which the staff reviews in Section 3G of this report).

Based on the above review, the staff finds that the applicant’s design and analysis procedures are rational; follow best practices in analytical modeling; captured all the containment internal structures in the IRB model to ensure appropriate vertical and lateral load transfer between connected components; and acceptable to meet SRP Section 3.8.3.II.4 acceptance criteria.

3.8.3.3.5 Structural Acceptance Criteria

The staff reviewed PSAR Section 3.8.3.5, as amended, related to structural acceptance criteria for containment internal structures, against the acceptance criteria in SRP Section 3.8.3.II.5, and noted the following:

- Acceptance criteria for the design of DP-SC RPV pedestal, including welded and bolted connections, are in accordance with Section 5.0 of NEDC-33926P-A.
- Acceptance criteria for the containment internal structural steel are in accordance with ANSI/AISC N690, including the guidance of RG 1.243.

The staff finds the applicant's structural acceptance criteria for the RPV pedestal of DP-SC construction acceptable because it is consistent with NRC-approved LTR NEDC-33926P-A, subject to CP Conditions L-CP-3.8-1 and L-CP-3.8-2 in Section 3.8.1.3.2 of this report. The staff also finds the applicant's acceptance criteria for containment internal structural steel acceptable because they are consistent with design code ANSI/AISC N690-18 as endorsed by RG 1.243.

3.8.3.3.6 Materials, Quality Control, and Special Construction Techniques

The staff reviewed PSAR Section 3.8.3.6 against the acceptance criteria in SRP Section 3.8.3.II.6, which states that the specified materials of construction and quality control programs are acceptable if found to be in accordance with the public code or standards used for the containment internal structures.

3.8.3.3.6.1 *Materials*

The staff reviewed PSAR Section 3.8.3.6.1 regarding materials of construction used for the containment internal structures and noted the following:

- High-density self-consolidating concrete infill designed for shielding in accordance with RG 1.69 is used in the RPV pedestal DP-SC modules, and design compressive strength is determined per Section 5.2.1 of LTR NEDC-33926P. Aggregates used in areas subjected to elevated temperature for long durations are free of pyretic materials.
- The RPV pedestal DP-SC steel components conform to the following:
  - Steel faceplates/diaphragm plate: ASTM A572 Grade 50 or Grade 65
  - Studs: ASTM A108 Type B
- Structural steel used in containment internal structures conform to
  - Structural steel and connections: ASTM A500 or ASTM A1085
  - High-strength structural steel plates: ASTM A572
  - Structural steel attachments to SCCV: ASTM A572 Grade 50 per ASME Code Case N-632
- Corrosion protection coatings are applied on the exposed surface of the containment internal structures to protect the metal from corrosion.

Based on the above review, the staff finds that the materials of construction used for the DP-SC RPV pedestal and containment internal structural steel components are acceptable because they are based on NRC-approved LTR, the design code ANSI/AISC N690-18 endorsed in RG 1.243, and recognized ASTM standards, and therefore meet the SRP Section 3.8.3.II.6 acceptance criteria.

3.8.3.3.6.2 *Quality Control*

The PSAR states that the quality control procedures governing the design, fabrication, construction, installation, and inspection of the DP-SC RPV pedestal are the same as those for the RB structure described in Section 3.8.4.1.6, which in turn states that the procedures are in accordance with Sections 5.16 and 5.17 of LTR NEDC-33926P. The staff finds this acceptable because the quality control procedures are in accordance with NRC-approved LTR NEDC-33926P, and therefore acceptable to meet the acceptance criteria in SRP Section 3.8.3.II.6.

The PSAR section also states that quality control procedures governing the design, fabrication, construction, installation, and inspection of the structural steel components of containment internal structures are in accordance with RG 1.28 and Chapters NM and NN of ANSI/AISC N690. The staff finds applicant's quality control procedures for structural steel will be consistent with RG 1.28 and the design and construction code ANSI/AISC N690-18 endorsed in RG 1.243, and therefore meet the acceptance criteria in SRP Section 3.8.3.II.6.

3.8.3.3.6.2.1 Special Construction Techniques

In PSAR Section 3.8.3.6.3, the applicant states that the RPV pedestal is constructed using modular construction technique as described in PSAR Section 3.8.4.1.6, which states that DP-SC modules are used that are fabricated and constructed following the requirements of Section 5.16 of LTR NEDC-33926P-A.

The staff finds the use of DP-SC modular construction as a special construction technique for the RPV pedestal part of the containment internal structures acceptable because their fabrication and construction follow the requirements of NRC-approved LTR NEDC-33926P-A.

3.8.3.3.7 Testing and In-Service Surveillance Requirements

The staff reviewed PSAR Section 3.8.3.7 against the acceptance criteria in SRP Section 3.8.3.II.7, which states that for seismic Category 1 structures inside containment accommodate in-service inspection of critical areas, and that structure monitoring and maintenance requirements are acceptable if found to be in accordance with 10 CFR 50.65 and RG 1.160.

Based on its review of the PSAR section, the staff noted the following:

- The condition of containment internal structures is monitored during the operating life of the plant in compliance with 10 CFR 50.65 and in accordance with RG 1.160, *Monitoring the Effectiveness of Maintenance in Nuclear Power Plants*.
- The containment internal structure design provides sufficient physical access to the structures to accommodate the ISI of the containment internal structures.

The staff noted from the descriptions of the containment internal structures evaluated in Section 3.8.3.1 of this report that the applicant has included provisions for providing physical access to the containment internal structures for inspection. The staff finds that the applicant's proposal for monitoring the condition of containment internal structures is in accordance with the maintenance rules and guidance in RG 1.160 and is acceptable because it is consistent with the acceptance criteria in SRP Section 3.8.3.II.7 for Category 1 structures inside containment.

3.8.3.4 Based on the staff review in Sections 3.8.3.3.2 through 3.8.3.3.7 of this report above, the NRC staff finds that the PSAR has adequately addressed the design bases and design criteria for the CIS and their relation of the design bases to the principal design criteria, applicable to CIS, by meeting the SRP 3.8.3 acceptance criteria;

thereby meeting the requirements of 50.34(a)(3)(ii). Additionally, based on the staff evaluation documented in Section 3G of this report of the preliminary analysis and evaluation results for the CIS in PSAR Appendix 3G, the staff finds that the requirements of 10 CFR 50.34(a)(4) are met.

3.8.3.5 Conclusion

The staff has reviewed the available information provided in the PSAR Section 3.8.3 and Appendix 3G, and, for the reasons given above, concludes that the design basis of the containment internal structures (RPV pedestal, CEPSS, support platforms) is acceptable, subject to satisfying CPA Conditions L-CP-3.8-1 and L-CP-3.8-2 in Section 3.8.1.3.2 of this report, to meet the relevant requirements of 10 CFR 50.55a; GDC 1, 2, 4, and 50; 10 CFR 50, Appendix B; and 10 CFR 50.65 by using RGs 1.69, 1.28, 1.243, and 1.160 and industry standards ANSI/AISC N690-18 and NRC-approved LTR NEDC-33926P-A to provide reasonable assurance that the containment internal structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Therefore, the NRC staff finds that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 50.34(a)(2), (3)(ii), (a)(3)(iii) and (a)(4) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35. The staff will confirm that the final containment internal structure design conforms to the design bases during the evaluation of the CRN-1 FSAR.

3.8.4 Other Seismic Category I Structures

3.8.4.1 Introduction

PSAR Section 3.8.4 describes the design basis criteria focused on the reactor building (RB) structure, the only seismic Category I structure other than containment for CRN-1. Other structures discussed, that are not seismic Category I with level of detail commensurate with the safety classification, include the RWB, which houses systems that process liquid, solid, and gaseous radioactive waste, and the control building (CB) locally hardened by design to meet the control room habitability requirements.

The applicant amended PSAR Section 3.8.3 by CPA Supplement 5, dated January 7, 2026 (TVA 2026-TN13029), and CPA Supplement 6, dated February 4, 2026 (TVA 2026-TN13031).

3.8.4.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (2), requires applicants to provide a summary description and discussion of the facility, with special attention to unusual or novel design features (in this case, steel-plate composite IRB using DP-SC modules) and principal safety considerations.
- 10 CFR 50.34(a), "Preliminary safety analysis report," sub-paragraph (3)(ii), requires applicants to address the design bases (in this case, for other seismic Category I structures) and the relation of the design bases to the principal design criteria, applicable to other seismic Category I structures, and sub-paragraph (3)(iii) requires applicants to provide information relative to materials of construction, general arrangement, and approximate

dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin of safety.

- 10 CFR 50.34(a)(4) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures (in this case, other seismic Category I structures) and transient conditions anticipated during the life of the facility, and the adequacy of structures and components (in this case, other seismic Category I structures, i.e., the IRB) provided for the prevention of accidents and the mitigation of the consequences of accidents.
- 10 CFR 50.55a and GDC 1 require that seismic Category I structures be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDC 2 requires that the seismic Category I structures be designed to withstand the most severe natural phenomena, such as winds, tornadoes, floods, and earthquakes, and the appropriate combination of all loads without loss of capability to perform their safety functions.
- GDC 4 requires that the seismic Category I structures be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- GDC 60, Control of Releases of Radioactive Materials to the Environment, requires that the nuclear power unit design include the means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operation, including anticipated operational occurrences. The release of radioactive materials from external man-induced events and design-basis accidents must also be controlled. GDC 60 applicable to the RWB is complied with through the use of RG 1.143 referenced below.
- 10 CFR Part 50, Appendix B, “Quality assurance criteria for nuclear power plants,” requires every applicant for a construction permit, to include in its preliminary safety analysis report, a description of the QA program to be applied to the design, fabrication, construction, and testing of the safety-related SSCs of the facility.
- 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” paragraph (a)(1), requires licensees of a nuclear power plant to monitor the performance or condition of SSCs important to safety, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these SSCs are capable of performing their intended functions.
- 10 CFR 73.55, “Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage,” subparagraph (e)(5), “Bullet resisting physical barriers,” requires that the reactor control room, the central alarm station, and the location within which the last access control function for access to the protected area is performed, must be bullet-resisting.

The following NRC SRP and regulatory guides (RGs) and other guidance documents also contain relevant guidance to satisfy requirements for this review:

- SRP Section 3.8.4, Revision 4 (September 2013), “Other Seismic Category I Structures,” provides guidance for materials, loads, and load combinations; design and analysis procedures; construction; examination; and testing related to seismic Category I structures (in this case, the RB structures surrounding the SCCV) other than containment and containment internal structures.

- RG 1.243, Revision 0, *Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments*, describes a method acceptable to the NRC staff for compliance with regulations for the design, fabrication, and erection of safety-related steel structures and steel-plate composite walls for other than reactor vessels and containments (NRC 2021-TN13112). 10 CFR Part 50, Appendix A, GDC 1, 2, and 4; and 10 CFR Part 50, Appendix B, are applicable. This guide endorses, with exceptions and clarifications, the procedures and standards of the ANSI/AISC N690-18 code, including provisions for steel-plate composite walls in Appendix N9.
- RG 1.143, Revision 2 (November 2001), *Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants*, provides guidance to applicants for complying with NRC regulations in the design, construction, installation, and testing of the SSCs of radioactive waste management facilities in light water reactor (LWR) nuclear power plants (NRC 2001-TN1134). GDC 1, 2, and 60 and 10 CFR 50, Appendix B are applicable to NRC RG 1.143.
- RG 1.142, Revision 3, *Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)*, May 2020 (NRC 2020-TN13101).
- RG 1.199, Revision 1, *Anchoring Components and Structural Supports in Concrete*, April 2020 (NRC 2020-TN13099).
- RG 1.26, Revision 6, *Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants*; December 2021 (NRC 2021-TN12803).
- RG 1.28, Revision 6, *Quality Assurance Program Criteria (Design and Construction)*, September 2023 (NRC 2023-TN13105).
- RG 1.160, Revision 4, *Monitoring the Effectiveness of Maintenance in Nuclear Power Plants* (NRC 2018-TN7799).
- RG 1.244, Revision 0, *Control of Heavy Loads at Nuclear Facilities*, December 2021 (NRC 2021-TN13113).
- RG 1.117, Revision 2, *Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants*, July 2016 (NRC 2016-TN13104).
- RG 1.70, Revision 3, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants – LWR Edition*, November 1978 (NRC 1978-TN12879).
- DNRL-ISG-2022-01, “Safety Review of Light-Water Power Reactor Construction Permit Applications,” Interim Staff Guidance, October 2022 (NRC 2022-TN12894) provides clarifying and supplemental guidance to the SRP written for combined licenses when used for CP reviews.
- Licensing Topical Report (LTR) NEDC-33926P/NEDO-33926P-A, Revision 4, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building (RB) Structural Design” (proprietary/non-proprietary) (ADAMS Accession No. ML25351A085 package, ML25351A087 proprietary; GE Hitachi 2025-TN13024 public); NRC Safety Evaluation is included as the front matter of the LTR NEDC-33926P/NEDO-33926P-A version; NRC Safety Evaluation of the LTR is also available at ADAMS Accession No. ML25268A140 – Proprietary, NRC 2025-TN13032 – Public.

- LTR NEDO-33914-A, Revision 2, "BWRX-300 Advanced Civil Construction and Design Approach," (GE Hitachi 2022-TN13022).
- ASCE/SEI 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," American Society of Civil Engineers, 2016.
- ASME NOG-1 "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," 2020 edition.

3.8.4.3. Technical Evaluation

3.8.4.2.1 Reactor Building

The staff reviewed PSAR Section 3.8.4.1 against the acceptance criteria SRP 3.8.4.1.II and referenced regulatory guidance therein, which is one acceptable way of meeting the applicable regulations identified in the regulatory evaluation.

3.8.4.3.1.1 *Description of the Structure*

The staff reviewed the descriptive information of the reactor building (RB), as amended, and its functions against the acceptance criteria in SRP Section 3.8.4.1.II.1, which refers to corresponding criteria in RG 1.70, Section 3.8.3.1.

The staff reviewed the structural and functional description of the RB in PSAR Section 3.8.4.1.1 and noted the following:

- The RB is a deeply embedded cylindrical structure made of floors, walls, roof, and a common basemat of steel-plate composite modules using diaphragm plates (DP-SC, addressed in LTR NEDC-33926P-A) or conventional steel-plate composite, with structural steel members supporting the DP-SC roof.
- General arrangement and approximate dimensions of the RB structures are illustrated in PSAR Figures 1.2-2, 1.2-3, 3.8-1, and 3.8-8, and the code jurisdictional boundary is shown in PSAR Figure 3.8-2.
- The RB fuel and equipment pool provides for storage of new and spent fuel along with in-core components.
- The RB polar crane consists of an overhead bridge of two deep girders supporting a trolley with main and auxiliary hooks.
- The primary functions of RB are described.

The staff finds that the level of detail in the PSAR description of general arrangement, configuration, functions, and principal features of the IRB structure is adequate for a CP because it is consistent with the criteria in SRP Section 3.8.4.II.1.

Based on the above review in Section 3.8.4.3.1 of this report, the NRC staff finds the PSAR provided an adequate summary description and discussion of the BWRX-300 RB structures, with special focus on novel design features (in this case, the RB construction using DP-SC modules) and its principal safety functions, and in Section 3.8.4.3.1.6 of this report materials of construction, thereby satisfying the requirements in 10 CFR50.34(a)(2) and (a)(3)(iii).

3.8.4.3.1.2 *Applicable Codes, Standards, and Specifications*

The staff reviewed PSAR Section 3.8.4.1.2, as amended, against the acceptance criteria in SRP Section 3.8.4.II.2.

The staff finds that codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and ISI of the RB DP-SC structures are as listed in NRC-approved LTR NEDC-33926P-A, and therefore acceptable.

The staff also finds that the applicant has identified the appropriate code and standard (ANSI/AISC N690-18) for the structural steel components (including the polar crane) of the RB because it is endorsed in RG 1.243, Revision 3, and therefore is consistent with the acceptance criteria in SRP Section 3.8.4.II.2.

The staff further noted that, in addition to meeting the requirements of ANSI/AISC N690-18 and RG 1.243, the RB polar crane rail clips, runway girders, and support corbels are also designed to meet the crane loading requirements of ASCE/SEI 7-16 and Sections 4160 and 4460 of ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," 2020 edition. The staff finds the use certain provisions of these standards for the polar crane components to be acceptable because ASCE 7-16 is a recognized standard for minimum design loads and ASME NOG-1 is endorsed in NRC RG 1.244.

3.8.4.3.1.3 *Loads and Loading Combinations*

The staff reviewed PSAR Section 3.8.4.1.3 design loads and load combinations for the RB shown in PSAR Table 3.8-7 against the acceptance criteria in SRP Section 3.8.4.II.3.B applicable for steel and steel-plate composite structures, and RG 1.243.

The staff finds the loads for design of the RB shown in PSAR Table 3.8-7 acceptable because they are based on design code ANSI/AISC N690-18 endorsed in RG 1.243, they include loads such as accident pressure and thermal transient loads internal to the SCCV for design of RB components that are integrated with the SCCV, and seismic hydrodynamic loads are calculated as evaluated in Section 3.7.1 of this report; therefore, meets the SRP Section 3.8.4.II.3.B acceptance criteria.

The staff also finds that the design loads and load combinations meet the acceptance criteria in SRP Section 3.8.3.II.3.B because they are consistent with design code AISC/N690-18 for steel structures, as supplemented by the regulatory positions in NRC RG 1.243, which also applies to steel-plate composite/DP-SC structures.

3.8.4.3.1.4 *Design and Analysis Procedures*

The staff reviewed the design and analysis procedures for the RB in PSAR Section 3.8.4.1.4, as amended, against the acceptance criteria in SRP Section 3.8.4.II.4. However, the staff did not conduct for CP review a structural design audit indicated in SRP Section 3.8.4.II.4.F described in Appendix B to SRP Section 3.8.4 because the structural design of the RB structures is incomplete and in progress during the review of the CPA.

The staff reviewed the analysis methods in PSAR Section 3.8.4.1.4 and noted the following:

- The RB, containment, and containment internal structures are modeled and analyzed as one integrated structure, using ANSYS and ACS SASSI computer programs, to determine design structural demands from design loads and load combinations.

- The validation of the computer programs used is described in PSAR Appendix 3I, which the staff evaluated and found acceptable in Section 3I of this report.
- Four types of linear elastic analyses are performed on IRB structural models that have the same nodal and element configurations to determine design demands, thereby enabling demands to be combined on an element-by-element basis for applicable design load combinations:
  - 1-g static SSI analyses (gravity loads due to dead load, vertical fluid loads, below-grade static lateral earth pressure load, and surcharge lateral pressure loads from surrounding power block buildings)
  - Static and quasi-static analyses (live loads, crane loads, structural integrity test (SIT) and accident containment internal pressure loads including differential and RB sub-compartment loads, hydrostatic pressure loads on pool walls, extreme wind loads, rain and snow loads, seismic water sloshing and breathing mode pressure loads on pool walls, high energy line break, equipment and pipe reaction loads, post-accident internal flood loads, groundwater pressure loads on IRB mat foundation and below-ground exterior wall)
  - Thermal stress analyses (thermal from normal operating and design-basis accident conditions)
  - Seismic SSI analyses (dynamic)
- Detailed information of the analysis models, analysis cases, model requirements, and boundary conditions is described in PSAR Appendices 3B, 3C, 3E, and 3H, which the staff evaluated and found acceptable in Sections 3B, 3C, 3E, and 3H of this report.
- The integrated RB structural model is developed following the general FE modeling guidelines in PSAR Section 3.7.2.3 (evaluated by staff in Section 3.7.2.3.3 of this report and Section 5.1.1 of LTR NEDO-33914-A), adequately represents the RB structural configuration, and meets the requirements for mesh refinement and quality attributes.
- Appropriate material properties and seismic mass inertia properties are assigned as discussed in PSAR Section 3.7.2.3.
- Effective stiffness properties of DP-SC and conventional steel-plate composite structural elements and connections are computed and assigned per Sections 5.5 and 5.6 of LTR-NEDC-33926P-A.
- Effect of interaction of the deeply embedded IRB with the surrounding subgrade is incorporated by considering surrounding soil and rock as a layered half-space continuum with elastic spring elements used to model the interfaces between the RB and surrounding subgrade. The spring elements provide results used in the calculation of static and dynamic earth pressures.
- Appropriate geotechnical design input parameters are used for static and thermal analyses as described in PSAR Section 3.8.5.4.1, and dynamic subgrade properties developed for the seismic SSI analysis are described in PSAR Section 3.7.1.3, which the staff evaluated in Sections 3.8.5.3.4.1 and 3.7.1.3.3 of this report.
- The subgrade modeling assumptions for the deeply embedded RB are in accordance with Section 5.1.2 of LTR NEDO-33914-A.
- Validation of the earth pressure loads for the RB design is performed using the results of the foundation interface analysis performed in accordance with Sections 4 and 5.1.3 of LTR

NEDO-33914-A addressing L&C 8.2 of NEDO-33914-A A (which the staff evaluated and found acceptable in Section 2.5.4.4 of this report).

Based on the above review, the staff finds that the applicant's analysis procedures to determine structural demands are rational, follow best practices in analytical modeling using validated computer programs and appropriate model inputs, and captured all the significant IRB structural elements and the interaction of the deeply embedded RB with the surrounding subgrade. Further, these analysis procedures are consistent with methodologies in NRC-approved LTRs NEDO-33914-A and NEDP-33926P-A and adequately meet the SRP Section 3.8.3.II.4 acceptance criteria, and are therefore acceptable.

The staff reviewed the design evaluation method for the RB structures in PSAR Section 3.8.4.1.4 and noted the following:

- Results of the linear elastic SSI analysis for the IRB models are evaluated for controlling loads and load combinations to identify critical cross sections with maximum structural demands.
- The design of the DP-SC RB structures, including connections, is performed in accordance with Section 5.0 of LTR NEDC-33926P-A.
- The design of RB steel structures and conventional steel-plate composite RB structures is performed in accordance with ANSI/AISC N690-18 and RG 1.243.

Based on the above review, the staff finds that the applicant's design procedures for the RB DP-SC, steel-plate composite, and steel structures are rational and consistent with SRP Section 3.8.3.II.4 acceptance criteria because they are based on NRC-approved LTR NEDC-33926P-A and/or industry code ANSI/AISC N690-18 as endorsed in NRC RG 1.243.

3.8.4.3.1.5 *Structural Acceptance Criteria*

The staff reviewed PSAR Section 3.8.4.1.5, as amended, related to structural acceptance criteria for the RB structures, against the acceptance criteria in SRP Section 3.8.4.II.5, and noted the following:

- Acceptance criteria for the design of DP-SC, steel-plate composite, and steel RB structures, including welded and bolted connections, are in accordance with ANSI/AISC N690 and Section 5.0 of NEDC-33926P.
- Acceptance criteria serviceability considerations are in accordance with Chapter NL of ANSI/AISC N690.
- Seismic design criteria for the RB structures in limit state D (LS-D) are in accordance with ASCE/SEI 43-19, which ensure essentially elastic behavior.
- Criteria for evaluation of interaction of the RB with adjacent power block structures are as discussed in PSAR Sections 3.3.2.3 and 3.7.2.3.8, and criteria for foundation stability evaluations are as described in PSAR 3.8.5.4.3, which are evaluated by staff in corresponding sections of this report.
- Acceptance criteria for design-basis impulsive and impactive loads are in accordance with ANSI/AISC N690-18 and associated guidance in RG 1.243, and Section 5.8 of LTR NEDC-33926P-A.

The staff finds the applicant's structural acceptance criteria for the design and serviceability of RB structures acceptable because they are consistent with industry standard ANSI/AISC N690-18 endorsed in RG 1.243 and NRC-approved LTR NEDC-33926P-A (subject to CPA conditions L-CP-3.8-1 and L-CP-3.8-2 discussed in Section 3.8.1.3.2 of this report) and ensure essentially elastic behavior of the RB structures under design loads; therefore, the acceptance criteria are consistent with SRP 3.8.4.II.5. The staff also finds the applicant's criteria for interaction evaluation of the RB structures with power block structures and stability evaluation of foundations of the RB and power block structures acceptable because they have been evaluated and found acceptable by staff in Sections 3.3.2.3.3, 3.7.2.3.8, 3.8.5.3.4.2, and 3.8.5.3.4.3 of this report.

3.8.4.3.1.6 *Materials, Quality Control, and Special Construction Techniques*

3.8.4.3.1.6.1 Materials

The staff reviewed PSAR Section 3.8.4.1.6, as amended, regarding materials of construction used for the RB structures against the criteria in SRP Section 3.8.4.II.6.

The staff finds that the materials of construction used for the concrete infill, steel materials, welded and bolted connections, coatings, and corrosion prevention are acceptable because they are consistent with NRC-approved LTR NEDC-33926P-A (subject to CPA conditions L-CP-3.8-1 and L-CP-3.8-2 discussed in Section 3.8.1.3.2 of this report), related codes and standards (ACI 349-13 and AISC N690-18), and recognized ASTM standards; therefore, they meet the SRP Section 3.8.4.II.6 acceptance criteria.

The staff finds the materials proposed for pool liners, namely carbon steel (ASTM A516 Grade 70) with stainless steel clad (ASTM A264 or ASTM A240 Type 304L or stainless steel (ASTM A240 Type 304L), are acceptable because they are based on typical ASTM standards used for fuel pool liners and the stainless steel material will provide corrosion resistance and capability to withstand environmental conditions of the fuel pool.

3.8.4.3.1.6.2. Quality Control

The PSAR Section 3.8.4.1.6, as amended, states that the quality control procedures governing the fabrication, furnishing, and installation of the RB structural components are in accordance with the requirements of Section NA5, Chapters NM and NN, of ANSI/AISC N690; RG 1.28 as recommended by RG 1.243; and as discussed in Sections 5.16 and 5.17 of LTR NEDC-33926P-A. The PSAR also states that quality control procedures governing the design, fabrication, construction, installation, and inspection of the RB are developed in compliance with GVH QA program. The staff finds this acceptable because the quality control procedures are in accordance with applicable NRC RGs, standards endorsed therein, NRC-approved LTR NEDC-33926P-A, and GVH QA program and therefore meet the acceptance criteria in SRP Section 3.8.4.II.6.

3.8.4.3.1.6.3. Special Construction Techniques

PSAR Section 3.8.4.1.6, as amended, states that the IRB is built using modular construction technique using DP-SC modules that are fabricated and constructed following the requirements of Section 5.16 of LTR NEDC-33926P-A.

The staff finds the use of DP-SC modular construction as a special construction technique for the IRB modules acceptable because its fabrication and construction follows the requirements of NRC-approved LTR NEDC-33926P-A.

3.8.4.3.1.7 *Testing and In-Service Surveillance Requirements*

The staff reviewed PSAR Section 3.8.4.1.7, as amended, against the acceptance criteria in SRP Section 3.8.4.II.7, which states that seismic Category I structures accommodate in-service inspection of critical areas, and that structures monitoring and maintenance requirements are acceptable if found to be in accordance with 10 CFR 50.65 and RG 1.160. Based on its review of the PSAR section, the staff noted the following:

- The condition of IRB is monitored during the operating life of the plant in compliance with 10 CFR 50.65 and in accordance with RG 1.160, *Monitoring the Effectiveness of Maintenance in Nuclear Power Plants*.
- The aging management, ISI, and testing of the RB DP-SC structures, in accessible and inaccessible areas, are performed in accordance with Section 5.18 of LTR NEDC-33926P, and inspections and monitoring during construction and commissioning are performed as described in Section 3.2 of LTR NEDO-33914-A. The post-construction testing and surveillance programs for RB below grade wall and foundation, including monitoring ground water chemistry, settlement, and differential displacements, are per Sections 3.3 and 3.4 of NEDO-33914-A.

The staff finds that the applicant's proposal for monitoring the condition of RB structures in accordance with the maintenance rule and guidance in RG 1.160 is acceptable because it is consistent with the acceptance criteria in SRP Section 3.8.4.II.7 for other seismic Category I structures. The approaches for DP-SC structures are acceptable because they are in accordance with NRC-approved LTRs NEDC-33926P-A and NEDO-33914-A.

Based on the staff review in Sections 3.8.4.3.1.2 through 3.8.4.3.1.7 of this report above, the NRC staff finds that the PSAR has adequately addressed the design bases and design criteria for the RB and their relation of the design bases to the principal design criteria, applicable to the foundations, by meeting the SRP 3.8.4 acceptance criteria; thereby meeting the requirements of 50.34(a)(3)(ii). Additionally, based on the staff evaluation documented in Section 3H of this report of the preliminary analysis and evaluation results for the RB in PSAR Appendix 3H, the staff finds that the requirements of 10 CFR 50.34(a)(4) are also met for issuance of a CP.

3.8.4.3.1.8 *Conclusion for RB*

The staff has reviewed the available information provided in the PSAR Section 3.8.4.1 and Appendix 3H, as amended, and, for the reasons given above, concludes, subject to satisfying CP conditions L-CP-3.8-1 and L-CP-3.8-2 in Section 3.8.1.3.2 of this report, that the design-basis criteria for safety-related integrated reactor building structures (i.e., the IRB) are acceptable to meet the relevant requirements of 10 CFR 50.55a; GDC 1, 2, and 4; and 10 CFR 50, Appendix B by using RGs 1.28, 1.243, and 1.160; industry standard ANSI/AISC N690-18; and LTRs NEDC-33926P-A and NEDO-33914-A to provide reasonable assurance that the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Therefore, the NRC staff finds that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 50.34(a)(2), (a)(3)(ii), (a)(3)(iii) and (a)(4) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to 10 CFR 50.35. The staff will confirm that the final IRB design conforms to the design bases during the evaluation of the CRN-1 FSAR.

3.8.4.2.2 Intervening Structural Elements Between Distribution Systems and Structures

PSAR Section 3.8.4.2 describes the design basis for intervening structural elements between distribution systems or equipment supports and structures, such as intervening structural elements, attached to walls, floors, and ceilings of structures for cable trays; conduits; heating, ventilation, and air conditioning ducts; and equipment. The qualification of supports themselves of these distribution systems and equipment is addressed in PSAR Section 3.9.2 and evaluated in Section 3.9.2 of this report.

The staff reviewed PSAR Section 3.8.4.3.2 against applicable criteria in SRP Section 3.8.4.II and noted the following:

- Intervening structural elements are accounted for in the design of the structures, and the type and spacing of intervening elements are determined by rigidity and stress considerations of the distribution systems.
- The intervening structural elements are designed to withstand effects of seismic and dynamic loads to prevent collapse and interaction of the distribution systems with seismic Category I SSCs to meet the requirements of GDC 2 and GDC 4 in accordance with PSAR Section 3.7.3 evaluated in Section 3.7.3.3 of this report.
- The design of embedded load bearing steel elements in concrete structures will conform to the guidance of RG 1.199.

The staff finds that the design basis for the intervening structural elements between distribution systems and structures to be acceptable because they will be designed to appropriate standards and guidance to meet GDC 2 and 4 to prevent collapse and interaction between the distribution systems and seismic Category I SSCs, thereby meeting the requirements in 10 CFR 50.34(a)(3)(ii) and (a)(3)(iii) with level of detail commensurate with safety-significance for issuance of a CP.

3.8.4.2.3 Radioactive Waste Building

The staff reviewed PSAR Section 3.8.4.3 on the RWB against the acceptance criteria in SRP Section 3.8.4.II and RG 1.143. The staff notes that by conforming to the guidance in RG 1.143, GDC 1, 2, and 60 and 10 CFR 50, Appendix B are met for the RWB.

3.8.4.3.3.1 *Description of the Structure*

The staff reviewed PSAR Section 3.8.4.3.1 on the RWB against the acceptance criteria in SRP Section 3.8.4.II.1 and RG 1.143. From review of the PSAR section, the staff noted the following:

- The RWB houses process systems for the liquid waste management system, the solid waste management system, and a portion of the off-gas system. The PSAR refers to Section 1.2 for RWB location and general dimensions.
- The RWB structure consists of a reinforced concrete exterior shear wall system with concrete floor slabs and interior columns resting on a shallow mat foundation supported by drilled shafts.
- The RWB is categorized as an RW-IIa structure per RG 1.143.

- A structural gap is maintained between the RB and RWB to prevent physical interaction.

The staff finds that the level of detail in the PSAR description of general location and dimensions, function, and principal features of the RWB is adequate for a CP because it is consistent with the criteria in SRP Section 3.8.4.II.1. Based on the above review in Section 3.8.4.3.3.1 of this report, the NRC staff finds the PSAR provided an adequate summary description and discussion of the RWB structure and its principal functions, and in Section 3.8.4.3.3.6 of this report materials of construction, thereby satisfying the requirements in 10 CFR 50.34 (a)(3)(iii) for issuance of a CP.

3.8.4.3.3.2 *Applicable Codes, Standards, and Specifications*

The staff reviewed PSAR Section 3.8.4.3.2 and finds it acceptable because the RWB is designed in accordance with RG 1.143, RG 1.243, and RG 1.142 and the codes and standards specified therein (ANSI/AISC N690-18, ACI 349-13), which is consistent with the guidance in SRP Section 3.8.4.

3.8.4.3.3.3 *Loads and Loading Combinations*

The staff reviewed PSAR Section 3.8.4.3.3 and finds it acceptable because the loads and load combinations used in the design of the RWB are in accordance with RG 1.143, Tables 2, 3, and 4, which is consistent with the guidance in SRP Section 3.8.4.

3.8.4.3.3.4 *Design and Analysis Procedures*

The staff reviewed design and analysis procedures for the RWB in PSAR Section 3.8.4.3.4 and noted the following:

- The RWB is part of the combined model used to evaluate the SSSI effects of the surrounding power block structures on the IRB.
- The design and seismic interaction evaluation of the RWB is performed using the ASCE 4-16 two-step linear analysis approach: the input foundation acceleration response spectra for RWB design earthquake (½ SSE) is developed from the IRB SSI analyses of the combined models, which is used in a fixed-base quasi-static or response analysis to develop seismic demands for RWB design. The demands for the RWB foundation design are obtained directly from the SSI analysis of the IRB with the power block structures.
- The RWB design is performed using RGs and standards identified in Section 3.8.4.3.3.2 of this report above.
- The RWB interaction and stability evaluation is performed per PSAR Section 3.7.2.8.

The staff finds the design and analysis procedures for the RWB acceptable because they are consistent with applicable NRC guidance and industry codes and standards in in SRP Section 3.8.4.II and identified in Section 3.8.4.3.3.2 of this report above.

3.8.4.3.3.5 *Structural Acceptance Criteria*

The RWB structure is designed to meet the acceptance criteria outlined in ANSI/AISC N690, as endorsed and modified by RG 1.243, and ACI 349-13, as endorsed and modified by RG 1.142, following the guidance of SRP Section 3.8.4.

3.8.4.3.3.6 *Material, Quality Control, and Special Construction Techniques*

Materials used in the construction of the RWB are in accordance with ACI 349-13 and ANSI/AISC N690, following the guidance of SRP Section 3.8.4 and RG 1.143. The RWB quality control procedures meet the guidance of RG 1.26, *Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants*; RG 1.143, Position 7; and ANSI/ANS-55.6, "Liquid Radioactive Waste Processing System for Light-Water Reactor Plants."

The staff finds the material and quality control aspects of the RWB acceptable because they are in accordance with the appropriate regulatory guidance and applicable codes and standards identified above. The RWB does not use special construction techniques, and therefore, no further staff review is required.

3.8.4.3.3.7 *Testing and In-Service Surveillance Requirements*

The staff noted the PSAR section refers to Chapter 11 for the testing and in-service requirements for radioactive waste process systems inside the RWB, which the staff evaluated in Sections 11.2, 11.3, and 11.4 of this report.

Based on the staff review in Sections 3.8.4.3.3.2 through 3.8.4.3.3.7 of this report above, the NRC staff finds that the PSAR has adequately addressed the design bases and design criteria for the RWB and their relation of the design bases to the principal design criteria, applicable to the RWB, by meeting the applicable SRP 3.8.4 acceptance criteria; thereby meeting the requirements of 50.34(a)(3)(ii) for issuance of a CP.

3.8.4.1.3.8 Conclusion for RWB

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the design-basis criteria for seismic Category RW-IIa RWB are acceptable to meet the relevant requirements of GDC 1, 2, 4, 60, and 10 CFR 50, Appendix B by using RGs 1.26, 1.243,1.142, and 1.143 and industry standards ACI 349-13 and ANSI/AISC N690-18 to provide reasonable assurance that the RWB will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Therefore, the NRC staff finds that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 50.34(a)(ii) and (a)(iii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.8.4.2.4 Augmented Power Block Structures

3.8.4.3.4.1 *Control Building (CB)*

The staff reviewed PSAR Section 3.8.4.4.1 against applicable criteria of SRP Section 3.8.4.II and noted the following:

- The seismic Category II CB structure houses the main control room (MCR) and related equipment and is constructed of reinforced concrete walls with steel columns and beams and composite floors, on a near surface concrete mat foundation supported by drilled shafts.
- The general arrangement of the seismic Category II CB structure is shown in PSAR Figures 1.2-2 through 1.2-7.

- The seismic Category II CB structure is designed to International Building Code (IBC) 2021 edition, and evaluated against limit state (LS) C criteria for seismic SSE and extreme wind event using approaches in Section 6 of LTR NEDO-33914-A.
- The CB, MCR, and egress route to the secondary control room (SCR) are evaluated to maintain structural integrity under SSE and extreme wind events per RG 1.117, and evaluated for seismic and extreme wind interaction with the IRB to meet habitability and safety requirements for MCR occupants and prevent adverse interaction with seismic Category I SSCs. The MCR and egress path are hardened for missiles from extreme wind events.
- The MCR is also designed to meet bullet resistant requirements of 10 CFR 73.55(e)(5), and MCR doors are to remain functional during and after a design-basis extreme event and ensure safe egress of MCR occupants to the secondary control room (SCR).

The staff finds the design basis description of the seismic Category II CB structure in the PSAR acceptable because it is based on a recognized building code, RG 1.117, and NRC-approved LTR, and adequate to prevent interaction with seismic Category I SSCs and to ensure the CB remains functional and the safety of MCR occupants during and after a design-basis extreme event. Based on the staff review above, the NRC staff finds that the PSAR has provided an adequate summary description and principal functions for the seismic category II CB commensurate with safety-significance, and adequately addressed the design bases and design criteria for the CB and their relation of the design bases to the principal design criteria, applicable to the CB, by meeting the applicable SRP 3.8.4.II acceptance criteria; and thereby meeting the requirements of 50.34(a)(3)(ii) and (a)(3)(iii) for issuance of a CP.

3.8.4.3.4.1.1 *Conclusion for CB*

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the design-basis criteria for seismic Category II CB are acceptable to meet the relevant requirements of 10 CFR 73.55(e)(5), industry standards IBC-2021, and NEDO-33914-A LTR to provide reasonable assurance that the RWB will withstand the specified design conditions without impairment of structural integrity to prevent interaction with seismic Category 1 SSCs and ensure safety of MCR occupants. Therefore, the NRC staff finds that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and (a)(3)(iii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.8.5 Foundations

3.8.5.1 Introduction

PSAR Section 3.8.5 describes general design rules for the seismic Category I DP-SC common mat foundation supporting the IRB. The PSAR section also includes discussion of adjacent non-seismic Category I power block structures' foundations due to their importance to the stability and structural integrity of the IRB.

The applicant amended PSAR Section 3.8.3 by CPA Supplement 5, dated January 7, 2026 (TVA 2026-TN13029).

3.8.5.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (2), requires applicants to provide a summary description and discussion of the facility, with special attention to unusual or novel design features (in this case, steel-plate composite IRB mat foundation) and principal safety considerations.
- 10 CFR 50.34(a), "Preliminary safety analysis report," sub-paragraph (3)(ii), requires applicants to address the design bases (in this case, for the IRB mat foundation) and the relation of the design bases to the principal design criteria, applicable to seismic Category I foundations, and sub-paragraph (3)(iii) requires applicants to provide information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin of safety.
- 10 CFR 50.34(a)(4) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures (in this case, IRB mat foundation) and transient conditions anticipated during the life of the facility, and the adequacy of structures and components (in this case, IRB mat foundation) provided for the prevention of accidents and the mitigation of the consequences of accidents.
- GDC 1 requires that seismic Category I structures including foundations (IRB mat foundation) be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDC 2 requires that the seismic Category I structures be designed to withstand the most severe natural phenomena, such as winds, tornadoes, floods, and earthquakes, and the appropriate combination of all loads without loss of capability to perform their safety functions.
- GDC 4 requires that the seismic Category I structures be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- GDC 50 requires that the SCCV containment foundation be designed with sufficient margin of safety to accommodate appropriate design loads.
- 10 CFR Part 50, Appendix B, "Quality assurance criteria for nuclear power plants," requires every applicant for a construction permit, to include in its preliminary safety analysis report, a description of the QA program to be applied to the design, fabrication, construction, and testing of the safety-related SSCs of the facility.
- 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," paragraph (a)(1), requires licensees of a nuclear power plant to monitor the performance or condition of SSCs important to safety, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these SSCs are capable of performing their intended functions.

The following NRC SRP and regulatory guides (RGs) and other guidance documents also contain relevant guidance to satisfy requirements for this review:

- SRP Section 3.8.5, Revision 4 (September 2013), “Foundations,” provides guidance for materials, loads and load combinations, design and analysis procedures, construction, examination, and testing related to foundations of seismic Category I structures.
- RG 1.243, Revision 0 (NRC 2021-TN13112), *Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments*, describes a method acceptable to the NRC staff for compliance with regulations for the design, fabrication, and erection of safety-related steel structures and steel-plate composite walls for other than reactor vessels and containments. 10 CFR Part 50, Appendix A, GDC 1, 2, and 4 and 10 CFR Part 50, Appendix B are applicable. This guide endorses, with exceptions and clarifications, the procedures and standards of the ANSI/AISC N690-18 code, including provisions for steel-plate composite walls in Appendix N9.
- RG 1.136, Revision 4, *Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments*, describes an approach that is acceptable to the NRC staff to meet regulatory requirements for materials, design, construction, fabrication, examination, and testing of concrete (reinforced or prestressed) containments in nuclear power plants (NRC 2021-TN13098). GDC 1, 2, 4, 16, and 50 are applicable to NRC RG 1.136.
- RG 1.28, Revision 6, *Quality Assurance Program Criteria (Design and Construction)*, September 2023 (NRC 2023-TN13105).
- RG 1.54, Revision 3, *Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants*, April 2017 (NRC 2010-TN8358). [Referenced in LTR NEDC-33926P-A]
- RG 1.160, Revision 4, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* (NRC 2018-TN7799).
- RG 1.70, Revision 3, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants – LWR Edition*, November 1978 (NRC 1978-TN12879).
- DNRL-ISG-2022-01, “Safety Review of Light-Water Power Reactor Construction Permit Applications,” Interim Staff Guidance, October 2022 (NRC 2022-TN12894) provides clarifying and supplemental guidance to the SRP written for combined licenses when used for CP reviews.
- ASCE/SEI 43-19, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, American Society of Civil Engineers, 2019.
- Licensing Topical Report (LTR) NEDC-33926P/NEDO-33926-A, Revision 4, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building (RB) Structural Design” (proprietary/non-proprietary) (ADAMS Accession No. ML25351A085 package, ML25351A087 proprietary; GE Hitachi 2025-TN13024 public); NRC Safety Evaluation is included as the front matter of the LTR NEDC-33926P/NEDO-33926P-A version; NRC Safety Evaluation of the LTR is also available at ADAMS Accession No. ML25268A140 – Proprietary, NRC 2025-TN13032 – Public.
- LTR NEDO-33914-A, Revision 2, “BWRX-300 Advanced Civil Construction and Design Approach,” (GE Hitachi 2022-TN13022).

3.8.5.3 Technical Evaluation

3.8.5.3.1 Description of the Foundations

The staff reviewed the descriptive information of the CRN-1 foundations against the acceptance criteria in SRP Section 3.8.5.II.1, which refers to corresponding criteria in RG 1.70, Section 3.8.5.1.

3.8.5.3.1.1 *RB and Containment Common Mat Foundation*

The staff reviewed PSAR Section 3.8.5.1.1 and noted the following:

- The IRB structure is founded on a deeply embedded DP-SC common mat foundation that supports the containment, containment internal structures, and RB.
- The general arrangement and dimensions of the CRN-1 seismic Category I IRB common mat foundation is illustrated in PSAR Figures 1.2-2, 1.2-3, 3.8-1, and 3.8-8.
- The containment boundary for the mat foundation extending to the outer perimeter of the SCCV is referred to as the inner mat foundation, and the portion outside the containment boundary is referred to as the outer mat foundation.

The staff finds that the level of detail in the PSAR description of general arrangement, configuration, overall dimensions, function and principal features of the IRB common mat foundation is adequate for a CP because it is consistent with the criteria in SRP Section 3.8.5.II.1.

3.8.5.3.1.2 *Other Power Block Foundations*

The staff reviewed PSAR Section 3.8.5.1.2 and noted the following:

- The RWB, turbine building (TB), service building, and reactor auxiliary structures are supported near the ground surface by drilled shaft deep foundations assigned the same seismic category as the supported superstructures. These foundations are separated from the RB exterior wall with a gap that can accommodate structural displacements due to seismic and extreme wind loads without physical interaction.
- The seismic Category RW-IIa RWB foundation is designed in accordance with ACI 349-13. The seismic Category II CB, TB, service building, and reactor auxiliary structure foundations are designed in accordance with International Building Code (2021 edition) and ACI 318-22 [ACI 318-19 Reissued 2022] invoked therein.
- Stability evaluations of the power block structures against sliding and overturning due to SSE and extreme wind and against flotation due to flooding are performed to ensure that stability requirements under normal and accident conditions are met. The seismic stability evaluations are performed using the results of seismic SSI analyses (discussed in PSAR Sections 3.7.2 and 3.8.4.1.4).
- The liquefaction potential, bearing capacity, and ground water effects of in-situ soil materials supporting the PBS foundations are described in PSAR Sections 2.5.4.8, 2.5.4.10, and 2.5.4.13. Staff evaluations of these are provided in Sections 2.5.4.8, 2.5.4.10, and 2.5.4.13 of this report.

The staff finds that the level of detail in the PSAR description of general arrangement, configuration, function, and principal features of the PBS foundations is adequate for a CP because, commensurate with their importance to safety, it is consistent with the criteria in SRP Section 3.8.5.II.1. Based on the above review in Section 3.8.5.3.1 of this report, the NRC staff finds the PSAR provided an adequate summary description and discussion of the BWRX-300 foundations for RB and containment and other power block structures, with special focus on novel design features (in this case, the RB foundation construction using DP-SC modules) and its principal safety functions, and in Section 3.8.5.3.3 of this report materials of construction, thereby satisfying the requirements in 10CFR50.34(a)(2) and (a)(3)(iii).

3.8.5.3.2 *Applicable Codes, Standards, and Specifications*

The staff reviewed PSAR Section 3.8.5.2, as amended, on applicable codes and standards for materials, design, fabrication, erection/construction, examination, testing, and in-service inspection of the IRB common mat foundation against the acceptance criteria in SRP Section 3.8.5.1.II.2. From review of the PSAR section, the staff noted the following:

- Applicable codes, standards, and specifications for the containment and RB common mat foundation are the same as those for the superstructures discussed in PSAR Sections 3.8.1.2 and 3.8.4.1.2. Staff evaluation of these applicable codes and standards is provided in Sections 3.8.1.3 and 3.8.4.3 of this report, respectively.
- Code jurisdictional boundary for application of the inner mat foundation (SCCV portion) is Section 6.0 of LTR NEDC-33926P-A and for the outer mat foundation is the modified N690-18 provisions in Section 5.0 of NEDC-33926P-A, as shown in PSAR Figure 3.8-2.

The staff finds that the codes and standards identified in the PSAR for the inner and outer mat foundation of the IRB common mat foundation meet the acceptance criteria in SRP Section 3.8.5.1.II.2 because they are the same as those for the corresponding superstructures, which the staff found acceptable in Sections 3.8.1.3 and 3.8.4.3 of this report.

3.8.5.3.3 *Loads and Loading Combinations*

The staff reviewed PSAR Section 3.8.5.3 against the acceptance criteria in SRP Section 3.8.5.II.3, which states the loads and load combinations are acceptable if found to be in accordance with those described in SRP Sections 3.8.1.II.3 for containment foundation and 3.8.4.II.3 for RB foundation (i.e., if they are the same as those for the respective superstructures). From review of PSAR Section 3.8.5.3, the staff noted the following:

- Design loads and load combinations for the containment and RB common mat foundation are those of the superstructures described in Sections 3.8.1.3 and 3.8.4.1.3.
- PSAR Section 3.8.1.3.6 states the load combinations and load factors for SCCV containment design are presented in PSAR Table 3.8-1 and are in accordance with ASME Code, Section III, Division 2, CC-3230, as supplemented by RG 1.136. The staff further notes that this is consistent with Section 6.0 of NRC-approved LTR NEDC-33926P-A for SCCV.
- PSAR Section 3.8.4.1.3 states that load combinations and load factors for RB design are presented in PSAR Table 3.8-7 and are in accordance with the provisions of Section NB2.5 of ANSI/AISC N690, as supplemented by regulatory positions 2.1 and 2.2 of RG 1.243. The staff further notes that this is consistent with Section 5.0 of NRC-approved LTR NEDC-33926P-A for RB structural design.

- For foundation stability against flotation, the site-specific design-basis flood is considered in combination with the dead load in accordance with SRP Section 3.8.5, "Foundations."

Based on the above review, the staff finds that the design loads and load combinations for the containment and RB common basemat foundation meet the acceptance criteria in SRP Section 3.8.5.II.3 because they are consistent with those of the respective superstructures, which are in accordance with the respective design codes, as endorsed in RGs 1.136 and 1.243 and NRC-approved LTR NEDC-33926P-A. The staff also finds the load combination for stability evaluation against flotation acceptable because it is consistent with that specified in SRP Section 3.8.5.II.3.

3.8.5.3.4 Design and Analysis Procedures

The staff reviewed the design and analysis procedures for the containment and RB common mat foundation in PSAR Section 3.8.5.4, as amended, against the acceptance criteria in SRP Section 3.8.5.II.4. However, the staff did not conduct for CP review a structural design audit indicated in SRP Section 3.8.5.II.4.G and described in Appendix B to SRP Section 3.8.4 because the structural design of the foundations is incomplete and in progress during the review of the CPA.

Based on review of PSAR Section 3.8.5.4, the staff noted the following on the design and analysis procedures for the DP-SC common mat foundation.

- The mat foundation is analyzed by linear elastic methods to determine the transfer of loads to the supporting media.
- Demands for the design are obtained from the one-step structural SSI analyses performed using the IRB model that include effects of interaction of the structure with surrounding subgrade and consider surcharge loads of the foundations of the surrounding power block structures, as described in PSAR Section 3.8.4.1.4.
- The DP-SC mat foundation is represented by thick shell elements. Subgrade properties used are evaluated in Section 3.7.1.3.3 of this report.
- Elastic springs are used to represent the stiffness properties of the foundation-subgrade interface. Vertical spring force results serve for calculations of foundation bearing stresses. Modeling methodology used to validate the subgrade pressure loads is based on Section 5.1.3 of NRC-approved LTR NEDO-33914-A.
- The containment (inner) mat foundation is designed in accordance with Section 6.0 of NRC-approved LTR NEDC-33926P-A. The outer mat of the common mat foundation is designed in accordance with Section 5.0 of LTR NEDC-33926P-A.

The staff finds that the design and analysis procedures for IRB common mat foundation are consistent with SRP Section 3.8.5.II.3 acceptance criteria because they are consistent with approaches in NRC-approved LTRs.

3.8.5.3.4.1 *Equivalent Linear Subgrade Properties*

The staff reviewed PSAR Sections 3.8.5.3.4.1 and noted the following:

- The 1-g static SSI analyses, subgrade impedance analyses, and thermal stress analyses use profiles of equivalent linear soil and rock properties developed using information from the existing CRN-1 site investigation and subsurface material testing programs summarized in Section 2.5.4. These properties are developed following Section 5.2.1 of NEDO-33914-A and consist of effective unit weight of soil materials below ground water table, elastic and shear moduli representing stiffness properties of soil and rock, and soil and rock Poisson's ratios representative of at-rest lateral pressure conditions.

- Two stratigraphy profiles, A and B, are developed to bound (lower and upper range) the geotechnical conditions across the site.
- The equivalent linear subgrade static profiles reflect anticipated as-built conditions at the site and include conservatisms.
- PSAR Table 3.8-8 provides a summary of the linearized static properties for the 1-g SSI and impedance analyses, and static properties for thermal analyses are provided in Table 3.8-9.

Based on the above review, the staff finds that the applicant's design and analysis procedures for developing equivalent linear subgrade static properties for the 1-g SSI analyses are acceptable because they are in accordance with NRC-approved LTR NEDO-33914-A and therefore adequately meet SRP Section 3.8.3.II.4 acceptance criteria. However, the staff did not conduct for CP review a structural design audit indicated in SRP Section 3.8.5.II.4.F described in Appendix B to SRP Section 3.8.4.

3.8.5.3.4.2 Soil Bearing Stability of Integrated Reactor Building Mat Foundation

The staff reviewed PSAR Section 3.8.5.3.4.1 and noted the following:

- The dynamic bearing pressure demands under SSE loads are calculated from the results of the seismic SSI analyses as described in Section 3.7.2.
- The static and dynamic stability of soil supporting the IRB mat foundation is evaluated in Section 2.5.4.10 per the guidance of SRP Section 2.5.4, "Stability of Subsurface Materials and Foundations."
- The bearing capacity of the rock supporting the IRB mat foundation at the CRN-1 site is provided in Section 2.5.4.10.1.

The analytical method for obtaining dynamic bearing pressure demands under SSE loads from the one-step seismic SSI analysis described in PSAR Section 3.7.2 was evaluated and found acceptable by staff in Section 3.7.2.3 of this report. For staff evaluation of the static and dynamic stability of soil supporting the IRB mat foundation and the bearing capacity of rock supporting the IRB mat foundation, refer to Section 2.5.4.10.3 of this report.

3.8.5.3.4.3 Stability of Power Block Foundations

The staff reviewed PSAR Section 3.8.5.4.3 and noted it states the following:

- The integrated RB mat foundation stability is assessed against sliding and overturning due to SSE and extreme wind, and flotation following the guidance of SRP Section 3.8.5.II.4.
- Explicit sliding and overturning stability evaluations are not performed for the deeply embedded RB because, in accordance with Section 7.2 of ASCE/SEI 43-19, its center of gravity (CG) is located deep below the grade elevation, and the structure is inherently stable against sliding and overturning.

- Required safety factors against sliding and overturning for the power block foundations under normal operating and accidental conditions are presented in Table 3.8-10, as amended.

The staff finds the method for assessing the IRB mat foundation stability against sliding and overturning due to extreme wind and flotation acceptable because it is stated to be consistent with the guidance in SRP Section 3.8.5.II.4.

Regarding the PSAR statement of not explicitly evaluating sliding and overturning stability of the deeply embedded IRB mat foundation based on the CG of the IRB being deep below grade, during the audit the staff requested the applicant to provide location coordinates of the CG of the IRB from the SSI analyses model to confirm the basis for the statement. The applicant indicated that the location coordinates of the CG were not readily available at this time and would be provided in the FSAR stage. This is tracked as operating license application (OLA) Action Item A-OL-3.8-1 described below**.** Nevertheless, based on the review of qualitative information of the embedment depth and distribution masses of the IRB from PSAR Figure 1.2-3, the staff finds that there is sufficient qualitative basis to make a reasonable judgment that the CG is expected to fall below grade, and therefore, not performing an explicit analysis would be acceptable. The staff will confirm this based on CG coordinates when made available during the FSAR review.

OL Action Item A-OL-3.8-1: TVA should provide in the FSAR the location coordinates of the CG of the IRB masses modeled in the SSI analysis model to confirm that the CG is located deep below the grade elevation as stated in PSAR Section 3.8.5.4.3.

The staff finds that the required safety factors against sliding and overturning for the power block structure provided in PSAR Table 3.8-10, as amended by CPA Supplement 5, (NRC 2026-TN13028 [package]), TVA 2026-TN13029 (PSAR update), are consistent with the criteria in SRP Section 3.8.5.II.5 for the stability load combinations and acceptable.

3.8.5.3.5 *Structural Acceptance Criteria*

The staff reviewed PSAR Section 3.8.5.5, related to structural acceptance criteria for containment and RB common mat foundation, against the acceptance criteria in SRP Section 3.7.5.II.5, and noted the PSAR states the following:

- The structural acceptance criteria for containment and RB common mat foundation are the same as those for their respective superstructures described in PSAR Sections 3.8.1.5 and 3.8.4.1.5, respectively.
- The PSAR refers to Sections 3.8.5.4.2 and 3.8.5.4.3, respectively, for safety factors considered for soil bearing pressure and foundation stability.

The staff finds the applicant’s structural acceptance criteria for the containment and RB common basemat foundation of DP-SC construction acceptable because they are respectively consistent with those of the superstructure in Sections 6.0 and 5.0 of NRC-approved LTR NEDC-33926P-A (subject to CPA conditions L-CP-3.8-1 and L-CP-3.8-2 discussed in Section 3.8.1.3.2 of this report). The staff evaluation of safety factors for soil bearing pressure and foundation stability is provided in Sections 3.8.5.3.4.2 and 3.8.5.3.4.3 of this report.

3.8.5.3.5 Materials, Quality Control, and Special Construction Techniques

The staff reviewed PSAR Section 3.8.5.6 against the acceptance criteria in SRP Section 3.7.5.II.6, which states that the specified materials of construction and quality control programs are acceptable if found to be in accordance with SRP Sections 3.8.1.II.6 and 3.8.5.I.6 for containment foundation and other Category I foundations, respectively.

3.8.5.3.6.1 *Materials*

The staff reviewed PSAR Section 3.8.5.6.1 regarding materials of construction used for the IRB common mat foundation and noted the following:

- Materials used for construction of the containment (inner) and RB (outer) common mat foundation are the same as those of the superstructures described in Sections 3.8.1.6.1 and 3.8.4.1.6, respectively.
- Common mat foundation DP-SC modules are designed with a sacrificial thickness not considered for strength to account for corrosion over the RB operating life, which is acceptable as a corrosion protection measure. The staff will review the determination of the sacrificial thickness when the justification information **CPA Condition 3.8-1** discussed in Section 3.8.1.3.2 of this report is provided by the applicant.

Based on the above review, the staff finds that the materials of construction used for the DP-SC containment (inner mat) and RB (outer mat) common mat foundation are acceptable because they are the same as those of the respective superstructures, which are respectively consistent with Sections 6.0 and 5.0 of NRC-approved LTR NEDC-33926P; therefore, they meet the SRP Section 3.7.5.II.6 acceptance criteria.

3.8.5.3.6.2 *Quality Control*

The staff reviewed PSAR Section 3.8.5.6.2 and noted that the quality control procedures governing the design, fabrication, construction, installation, and inspection of the DP-SC IRB common mat foundation are the same as those for the SCCV and RB structure described, respectively, in Sections 3.8.1.6.2 and 3.8.4.1.6, which are based on Sections 6.15 and 5.1, respectively, of LTR NEDC-33926P-A. By CPA Supplement 2 dated October 30, 2025 (TVA 2025-TN13046, TVA 2025-TN13027), the applicant also clarified in the amended PSAR Table 1.9-20 the applicability of RG 1.28 Rto PSAR Section 3.8.5.6. The staff finds this acceptable because the quality control procedures for the IRB inner and outer mat foundation are in accordance with NRC-approved LTR NEDC-33926P-A, which the staff found acceptable in Sections 6.15 and 5.17 of its SE (ML25268A140 – Proprietary, NRC 2025-TN13032 – Public) in LTR NEDC-33926P-A and conforms to RG 1.28; therefore, these procedures meet the acceptance criteria in SRP Section 3.8.5.II.6 and thereby meet the requirements of 10 CFR 50, Appendix B.

3.8.5.3.6.3 *Special Construction Techniques*

The staff reviewed PSAR Section 3.8.5.6.3 and noted the applicant refers to PSAR Section 3.8.4.1.6 for the DP-SC modular construction technique used for the IRB common mat foundation. PSAR Section 3.8.4.1.6 states that DP-SC modules are used that are fabricated and constructed following the requirements of Section 5.16 of LTR NEDC-33926P-A.

The staff finds the use of DP-SC modular construction as a special construction technique for the IRB common mat foundation acceptable because its fabrication and construction follows the requirements of NRC-approved LTR NEDC-33926P-A.

3.8.5.3.7 *Testing and In-Service Surveillance Requirements*

The staff reviewed PSAR Section 3.8.5.7, as amended, against the basic acceptance criteria in SRP Section 3.8.5.II.7, which states that, for seismic Category I foundations, structure monitoring and maintenance requirements are acceptable if found to be in accordance with 10 CFR 50.65 and RG 1.160. It also states that it is important to accommodate in-service inspection of critical areas. It further states that a foundation monitoring program includes monitoring of settlement (differential and total) during and post-construction to ensure the foundation continues to perform as designed.

Based on its review of PSAR Section 3.8.5.7, the staff noted the following:

- The inspection and testing of the integrated RB mat foundation follow the guidance of Sections 3.2.2 and 3.4 of NEDO-33914-A and Sections 5.18 and 6.22 of NEDC-33926P-A.
- The foundation testing and surveillance program complies with the requirements of 10 CFR 50.65 maintenance rule and guidance of RG 1.160 and includes the use of nondestructive testing and remote visual monitoring to monitor conditions in inaccessible areas of the foundation.
- Field instrumentation is installed inside and outside of the RB in accordance with Section 3.4 of NEDO-33914-A to measure the distribution of pore pressures around and below the structure, total settlement and tilt of the RB including foundation, and measurement of the surrounding power block structures.
- The instrumentation provides recordings during excavation and continuing through plant operation that can be benchmarked against design estimates using results of the foundation interface analysis (FIA).
- Short-term and long-term settlement monitoring plans are also developed to detect
  - vertical and horizontal movements in and around the integrated RB
  - differential distortion across the foundation footprint
  - differential settlements between the containment and RB portions of the mat foundation

Based on the above review, the staff finds that the applicant’s testing and surveillance program for the containment and RB common mat foundation includes condition monitoring and instrumented settlement monitoring (differential and total) during excavation and continuing into plant operation in accordance with the maintenance rule and guidance in RG 1.160 and the guidance in NRC-approved LTRs NEDO-33914-A and NEDC-33926P-A for DP-SC foundation, and is therefore adequate to meet the acceptance criteria in SRP Section 3.8.5.II.7.

Based on the staff review in Sections 3.8.5.3.2 through 3.8.5.3.7 of this report above, the NRC staff finds that the PSAR has adequately addressed the design bases and design criteria for the foundations of RB, containment and other power block structures, and their relation of the design bases to the principal design criteria, applicable to the foundations, by meeting the SRP 3.8.5 acceptance criteria; thereby meeting the requirements of 50.34(a)(3)(ii). Additionally, based on the staff evaluation documented in Section 3H of this report of the preliminary analysis and evaluation results for the RB including its foundation in PSAR Appendix 3H, the staff finds that the requirements of 10 CFR 50.34(a)(4) are also met.

3.8.5.4 Conclusion

The staff has reviewed the available information provided in the PSAR Section 3.8.5 and Appendix 3H, and, for the reasons given above, concludes that the design basis of the seismic Category I foundations is acceptable to meet the relevant requirements of 10 CFR 50.55a; 10 CFR Part 50, Appendix B; and GDC 1, 2, 4, and 50 by using RGs 1.136, 1.243, 1.28, and 1.160 and NRC-approved LTRs NEDC-33926P-A (subject to CPA conditions L-CP-3.8-1 and L-CP-3.8-2 discussed in Section 3.8.1.3.2 of this report) and NEDO-33194-A to provide reasonable assurance that seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions. Therefore, the NRC staff finds that the information provided by the applicant is sufficient to meet the regulatory requirements of 10 CFR 50.34(a)(2), (a)3(ii), (a)(3)(iii) and (a)(4) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to 10 CFR 50.35. The staff will confirm that the final design conforms to the design bases during the evaluation of the CRN-1 FSAR.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Introduction

This section addresses information concerning methods of analysis for seismic Category I components and supports, including those designated as ASME BPVC, Section III, Class 1, 2, 3, or core support structures.

3.9.1.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for methods of analysis for seismic Category I components and supports, including those designated as ASME BPVC, Section III, Class 1, 2, 3, or core support structures.
- In 10 CFR Part 50, Appendix A, GDC 1, the NRC requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- In 10 CFR Part 50, Appendix A, GDC 2, the NRC requires, in part, that SSCs important to safety be designed to withstand seismic events without loss of capability to perform their safety functions.
- In 10 CFR Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary" (RCBP) the NRC requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- In 10 CFR Part 50, Appendix A, GDC 15, “Reactor Coolant System Design,” the NRC requires that the reactor coolant system (RCS) and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).
- In 10 CFR Part 50, Appendix B, the NRC requires that QA criteria be provided for design control.
- In 10 CFR Part 50, Appendix S, the NRC requires that the plant design bases for mechanical components be established in consideration of site seismic characteristics.
- In 10 CFR 50.55a, the NRC endorsed ASME BPVC, Section III, Division 1 for the design of SSCs
- SRP Section 3.9.1, “Special Topics for Mechanical Components,” lists acceptance criteria adequate to meet the above requirements. The following reference document contains relevant guidance for this review: ASME BPVC, Section III.

3.9.1.3 Technical Evaluation

In this section of this report, the NRC staff reviews four topics: (1) design transients, (2) computer programs used in analyses, (3) experimental analysis, and (4) considerations for the evaluation of fault conditions.

Design Transients

PSAR, Section 3.9.1.1 states that the plant duty cycles represent transient events and their associated number of occurrences for the 60-year plant design life. These duty cycles are used for development of the BWRX-300 system and component design during normal operation, AOO, DBA, and testing. Requirements are evaluated for the system design and performance as they relate to complete reactor operation. The NRC staff finds that the process to define the transients and number of events for the design of risk significant SSCs is consistent with SRP 3.9.1. PSAR, Table 3.9-2, “Cycles of Events” lists the number of events for transients. The NRC staff will review the completeness of the list of transients and the appropriate number of cycles during the operating license application stage because this information is not required for issuance of a CP.

<u>*Computer Programs Used in Analyses*</u>

PSAR, Section 3.9.1.2, “Computer Programs Used in Analyses” states that the computer programs used in the analyses of the seismic Category I components are maintained by GVH or outside computer program developers. The GVH software is controlled under NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description.” The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature. The NRC staff finds that the verification methods discussed above for computer programs are consistent with SRP 3.9.1, and therefore acceptable. PSAR, Appendix 3J, “Computer Programs Used in the Design and Analysis of Structures, Systems and Components – Mechanical Design” lists computer programs used in the mechanical system and component analysis of seismic Category I components. Because this information is not required for issuance of a CP, the NRC will review the benchmarking of these computer programs during the operating license stage.

Experimental Stress Analysis

PSAR, Section 3.9.1.3, “Experimental Stress Analysis,” states that experimental stress analysis methods are used in compliance with the provisions of ASME BPVC, Section III, Appendices. The NRC staff finds that the use of ASME BPVC, Section III, Appendix II, “Experimental Stress Analysis and Determination of Stress Intensification Factors,” for experimental stress is consistent with SRP 3.9.1, and therefore acceptable.

Considerations for the Evaluation of Fault Conditions

Section 3.9.1.4, “Considerations for the Evaluation of Fault Conditions” states that the Service Level D (faulted conditions) design of pressure boundary components including piping is in accordance with ASME BPVC, Section III, Division 1, Subsection NB, Classes 2, and 3 piping per Subsection NCD, reactor vessel internals (RVI) per Subsection NG, and component supports per Subsection NF. The methods of analysis to calculate the stresses and deformations conforms to the methods outlined in the ASME BPVC, Section III, Mandatory Appendix XXVII, “Design by Analysis for Service Level D,” subject to the conditions addressed in SRP 3.9.1. Elastic analysis is performed according to the ASME BPVC. The NRC finds that the design and analysis discussed above is acceptable because 10 CFR 50.55a endorses ASME BPVC, Section III, Division 1, including the mandatory appendices for the design and analysis of nuclear components.

As discussed above, the design of the ASME BPVC, Section III, Class 1, 2, 3, or core support structures meet the provisions of SRP 3.9.1, and therefore, meet the relevant requirements of 10 CFR Part 50, Appendix B; 10 CFR Part 50, Appendix S; and GDC 1, 2, 14, and 15. As such, the design bases for methods of analysis for seismic Category I components and supports for the preliminary design have been explained sufficiently under 10 CFR 50.34(a)(3)(ii).

3.9.1.4 Conclusion

The staff has reviewed the available information provided in the PSAR, and, for the reasons given above, concludes that the applicant has met the relevant requirements 10 CFR Part 50, Appendix B; 10 CFR Part 50, Appendix S; and GDC 1, 2, 14, and 15. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.1 Introduction

This section provides the analytical methodologies, testing procedures, and dynamic analyses that the applicant used to ensure the structural and functional integrity of the piping systems, mechanical equipment, reactor vessel internals (RVIs), and their supports under vibratory loadings, including those caused by fluid flow and postulated seismic events.

This section addresses six main areas of review:
1. piping vibration, thermal expansion, and dynamic effects testing
2. seismic analysis and qualification of seismic Category I mechanical equipment
3. dynamic response analysis for RVIs under operational flow transients and steady-state conditions

4. preoperational flow-induced vibration (FIV) testing of RVIs
5. dynamic system analysis of the RVIs under faulted conditions
6. correlations of RVI vibration tests with the analytical results

3.9.2.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for addressing the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings, including those due to flow-induced excitations and postulated seismic events.
- In 10 CFR Part 50, Appendix A, GDC 1, the NRC requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- In 10 CFR Part 50, Appendix A, GDC 2, the NRC requires, in part, that SSCs, without loss of capability to perform their safety functions, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads, and to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.
- In 10 CFR Part 50, Appendix A, GDC 4, the NRC requires protection of SSCs against dynamic effects, including those of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- In 10 CFR Part 50, Appendix A, GDC 14, the NRC requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- In 10 CFR Part 50, Appendix A, GDC 15, the NRC requires, in part, that the RCS be designed with sufficient margin to assure that the RCPB is not exceeded during normal operation, including AOOs.
- In Appendix S to 10 CFR Part 50, the NRC requires that certain SSCs be designed to remain functional for an SSE.
- In Appendix B to 10 CFR Part 50, the NRC requires quality assurance in the dynamic testing and analysis of SSCs.

SRP Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," lists acceptance criteria adequate to meet the above requirements. The following NRC RG and reference document contain relevant guidance for this review.

- RG 1.20, Revision 4 (NRC 2017-TN13100), "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," as it relates to the vibration analysis and testing methodologies of the RVIs
- ASME Operation and Maintenance of Nuclear Power Plants Code (OM Code), Division 2, OM Standards, Part 3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems," and Part 7, "Requirements for Thermal

Expansion Testing of Nuclear Power Plant Piping Systems," as they relate to guidance for test specifications, as endorsed by SRP 3.9.2.

3.9.2.3 Technical Evaluation

This section of the SE, the NRC staff reviews the criteria, testing procedures, and dynamic analyses used to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor vessel internals (RVIs), and their supports under vibratory loadings, including those due to fluid flow and postulated seismic events.

3.9.2.3.1 Flow-Induced Vibration, Thermal Expansion, and Dynamic Effects

PSAR, Section 3.9.2.1, "Flow Induced Vibration, Thermal Expansion, and Dynamic Effects" states that piping vibration, thermal expansion, and dynamic effects testing for piping systems is performed in accordance with the startup administrative manual as described in Chapter 14 and follows guidance from ASME Operations and Maintenance of Nuclear Power Plants (OM) Code, Division 2, "OM Standards, Part 3 and Part 7. The NRC staff finds that using ASME OM Code, Part 3 and Part 7 for performing piping vibration, thermal expansion, and dynamic effects testing during startup testing is consistent with SRP 3.9.2, and therefore, acceptable. The applicant is updating the PSAR in response to Audit Question A-3.9-1, and this is being tracked as a **Confirmatory Item**.

3.9.2.3.2 Seismic Qualification of SC1 Mechanical Equipment (Including Other Reactor Building Vibration Induced Loads)

PSAR Section 3.9.2.2, "Seismic Qualification of SC1 Mechanical Equipment (Including Other Reactor Building Vibration Induced Loads)" refers to PSAR Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment" for the testing and analytical criteria for the seismic qualification of Seismic Category I mechanical equipment. The NRC staff's review is discussed in Section 3.10 of the report.

3.9.2.3.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

PSAR, Section 3.9.2.4, "Preoperational Flow-Induced Vibration Testing of Reactor Internals," states that the flow-induced vibration (FIV) program description has been developed in accordance with RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Startup Testing." Analysis is performed on components subject to operational flow transients and non-normal operational transient conditions. Vibration levels of each reactor internal components are scaled based on legacy BWR plant measurements. The analytical calculations will evaluate typical FIV mechanisms that occur in BWRs such as turbulence buffeting and vortex shedding. The dynamic analysis results of the reactor internals will generate the allowable vibration levels for use in the vibration measurement program during initial startup. The NRC staff finds that the plan to evaluate the FIV mechanisms and vibration level is consistent with RG 1.20 and analytic practice of past applicants, and therefore acceptable.

3.9.2.3.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

PSAR, Section 3.9.2.4, "Preoperational Flow-Induced Vibration Testing of Reactor Internals," states that the comprehensive vibration assessment program (CVAP) is used to verify the adequacy of the reactor vessel internals to withstand FIV and is developed in accordance with RG 1.20. The program establishes the necessary testing for the reactor internals to ensure that FIV experienced during normal operation will not cause structural failure or degradation.

Most of the BWRX-300 RVIs are classified as limited prototype or non-prototype per RG 1.20 as the components are scaled versions of components in existing legacy BWR plants. The BWRX-300 reactor internals, as a whole, are classified as prototype. Although the BWRX-300 steam dryer is a scaled version of the latest steam dryer technology, GVH plans a full measurement program during initial startup testing because of the new piping/orifice arrangement and different steam dome acoustics. The first operational BWRX-300 will follow the provisions of RG 1.20 to implement the measurement program during initial startup testing for all prototype and limited prototype components. The initial startup test series will verify the anticipated single- and two-phase flow on the vibration response of reactor internals and to ensure these effects are within acceptable FIV criteria developed from the dynamic analysis. The initial startup testing of reactor internals performed in accordance with the initial test program is discussed in PSAR, Chapter 14. Vibration measurements are recorded at incremental hold points up to 100 percent rate power during initial startup testing. The startup data will be recorded at plant steady state conditions at each hold point. Following the initial startup test program, reactor internal components will be inspected at the end of the first cycle to identify any indications of FIV per RG 1.20.

The NRC staff finds that the classification of the reactor internals and the vibration measurement and inspection plan during initial startup testing including the incremental hold points and evaluation of measurement data to verify the response of the reactor internals are within FIV acceptance criteria are consistent with RG 1.20, and therefore acceptable.

3.9.2.3.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

PSAR, Section 3.9.2.5, “Dynamic System Analysis of Reactor Internals Under Faulted Conditions,” states that all transients, including faulted conditions are analyzed to ASME BPVC service level definitions. PSAR, Table 3.9-1, “Design Condition Definition,” indicates that faulted condition is designed as ASME BPVC Service Level D. The dynamic analysis for faulted condition events considers loss-of-coolant-accident (LOCA) loads and seismic loads. The NRC staff finds that the ASME BPVC service level designation and the load combination for the faulted condition event are consistent with SRP 3.9.2, and therefore acceptable.

3.9.2.3.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

PSAR, Section 3.9.2.6, “Correlations of Reactor Internals Vibration Tests with the Analytical Results,” states that prior to initiation of the instrumented vibration measurement program, extensive dynamic analyses of the reactor internals are performed. The results of these analyses are used to generate the allowable vibration levels during the initial startup testing. The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis. Insight gained from the comparison is used to update the dynamic analysis models. The NRC staff finds that the plan to obtain initial startup testing measurement data and compare the test data to analytical results is consistent with RG 1.20 and therefore is acceptable.

As discussed above, PSAR Section 3.9.2 meets the guidance in SRP 3.9.2 and RG 1.20, and therefore, also meets the relevant requirements in Appendix S to 10 CFR Part 50, Appendix B to 10 CFR Part 50, 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 14, and 15. As such, the design bases for methods of dynamic testing and analysis of SSCs for the preliminary design have been explained sufficiently under 10 CFR 50.34(a)(3)(ii).

3.9.2.4 Conclusion

The staff has reviewed the available information provided in the PSAR, and, for the reasons given above, concludes that the applicant has met the relevant requirements of GDC 14 and 15 for the description of plan for design and testing of the piping systems, has met the requirements of GDC 1 and 4 for description of the plan for design and testing of reactor internals with the potential to generate loose parts to quality standards commensurate with the importance of the safety functions performed with appropriate protection against dynamic effects, has met the relevant requirements of GDC 2 and 4 for description of the plan of design of systems and components important to safety to withstand the effects of earthquakes and appropriate combinations, and has met the relevant requirements of GDC 1 for systems and components designed and tested to quality standards commensurate with the importance of the safety functions performed by the proposed program to correlate the test measurements with the analysis results. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.9.3 ASME Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Introduction

This subsection discusses the structural integrity and/or functional integrity requirements of pressure-retaining components, their supports, and core support structures that are designed in accordance with the rules of ASME BPVC, Section III.

3.9.3.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is for structural integrity and/or functional integrity requirements of pressure-retaining components, their supports, and core support structures that are designed in accordance with the rules of ASME BPVC, Section III.
- In 10 CFR Part 50.55a, the NRC requires that the design, fabrication, erection, and testing of SSCs be in accordance with the quality standards that are commensurate with the importance of the safety function to be performed.
- In 10 CFR Part 50, Appendix A, GDC 1, the NRC requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- In 10 CFR Part 50, Appendix A, GDC 2, the NRC requires, in part, that SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function.

- In 10 CFR Part 50, Appendix A, GDC 4, the NRC requires SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including lost-of-coolant accidents.
- In 10 CFR Part 50, Appendix A, GDC 14, the NRC requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- In 10 CFR Part 50, Appendix A, GDC 15, the NRC requires, in part, that the RCS and associated auxiliary, control and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during normal operation, including AOOs.
- In Appendix S to 10 CFR Part 50, the NRC requires that certain SSCs be designed to remain functional for an SSE.

SRP Section 3.9.3, “ASME Code Class 1, 2, and 3 Components and Component Support,” lists acceptance criteria adequate to meet the above requirements. ASME BPVC, Section III, provides relevant guidance for this review.

3.9.3.3 Technical Evaluation

In this section of the SE, the NRC staff reviews the structural integrity and/or functional integrity provisions of pressure-retaining components, their supports, and core support structures that are designed to ASME BPVC, Section III subsections.

3.9.3.3.1 Loading Combinations and Stress Limits

PSAR Section 3.9.3.1, “Loading Combinations, Design Transients, and Stress Limits,” provides criteria for selection and definition of design limits and loading combinations associated with normal operation, anticipated operational occurrences, design basis accident events, and seismic events for the design of ASME BPVC Section III components. PSAR Section 3.9.3.3, “Events Considered in Evaluating Effect of Loads on Fixed Equipment,” states that event combinations are divided into four plant conditions and ASME BPVC, Section III service levels as shown in PSAR table 3.9-1. Normal operation, anticipated operational occurrences, design basis accident events, and seismic events correspond to normal, upset, emergency, and faulted service conditions, respectively, and they in turn equate to ASME BPVC Service Levels A, B, C, and D, respectively. PSAR Section 3.9.3.1 states that specific load combinations for the ASME BPVC, Section III Class 1, 2, and 3 components, component supports, and core support structures are presented in Table 3.9-3, “Load Combinations and Acceptance Criteria.” PSAR Section 3.9.3.6.3, “Service Conditions” describes the component design analysis method to meet the stress limits for each ASME BPVC service level.

The NRC staff finds that the assignment of plant events to ASME BPVC service levels and the use of load combinations and stress limits for each service level for the ASME BPVC, Section III Class 1, 2, and 3 components, component supports, and core support structures is consistent with the guidance in SRP 3.9.3, including Appendix A, Table I, “Allowable Service Stress Limits for Specified Loading Combinations for ASME Section III Class 1, 2, and 3 Components, Component Support, and Core Support Structures.”

3.9.3.3.2 Pressure Relief Devices

PSAR Section 3.9.3.13, "Design of Pressure Relief Devices" states that the reactor coolant system (RCS) does not utilize safety or relief valves for overpressure relief. PSAR Section 6.3 describes the method of reactor pressure vessel overpressure relief that is based on NEDC-33910P-A, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection." During normal operation, the main steam flow to the turbines is throttled to control system pressure. The NRC staff finds that safety or relief valves are not applicable to the BWRX-300 design.

3.9.3.3.3 Component Supports

PSAR 3.9.3.14, "Component Supports" states that the establishment of the design/service loadings and limits is in accordance with ASME BPVC, Section III, Subsection NF, "Supports." These loadings and stress limits apply to the structural integrity of components and supports when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events. ASME BPVC, Section III component supports are designed, manufactured, installed, and tested in accordance with all applicable codes and standards. Supports and their attachments for ASME BPVC, Section III Class 1, 2, and 3 piping are designed in accordance with ASME BPVC, Section III, Subsection NF up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF. Pipe supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers and limit stops. The NRC staff finds that designing ASME BPVC, Section III Class 1, 2, and 3 component supports including service loading and stress limits to ASME BPVC, Section III, Subsection NF is consistent with SRP 3.9.3 and therefore, acceptable.

As discussed above, PSAR Section 3.9.3 meets the guidance in SRP 3.9.3, and therefore meets the relevant requirements of 10 CFR 50.55a, 10 CFR 50.34(a)(3)(ii), 10 CFR Part 50, Appendix S and GDC 1, 2, 4, 14, and 15. As such, the design bases for methods of design for ASME Class 1, 2, and 3 components, component supports, and core support structures for the preliminary design have been explained sufficiently under 10 CFR 50.34(a)(3)(ii).

3.9.3.4 Conclusion

The staff has reviewed the available information provided in the PSAR, and, for the reasons given above, concludes that the applicant's description of the plan to specify design and service combinations of loadings to ASME Code Class 1, 2, and 3 pressure retaining components is acceptable for CP issuance and meet the requirements of 10 CFR 50.55a, 10 CFR Part 50, Appendix S and GDC 1, 2, 4, 14, and 15. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.9.4 Control Rod Drive System

3.9.4.1 Introduction

The Control Rod Drive (CRD) system provides the primary means of reactivity control during normal, abnormal, and accident conditions in CRN-1. The CRD system design bases include two diverse motive forces for control rod insertion: (1) high-pressure water for SCRAM[2] and (2) fast motor insertion using fine motor control rod drive (FMCRD) motors. The CRN-1 design incorporates positioning and protective features that prevent inadvertent withdrawal, drop, and ejection of the control rod due to a component failure or other malfunction in accordance with the NRC regulatory requirements. PSAR Section 4.6, “Design of Reactivity Control Systems,” provides additional details regarding the CRD system for CRN-1.

3.9.4.2 Regulatory Evaluation

The NRC regulations in 10 CFR 50.34 (TN249), “Contents of applications; technical information,” in paragraph (a), *Preliminary safety analysis report*, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. Paragraph (2) in 10 CFR 50.34(a) requires that the PSAR provide a summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. Paragraph (3) in 10 CFR 50.34(a) requires in item (i) that the PSAR provide a preliminary design of the facility including the principal design criteria for the facility with reference to Appendix A, “General Design Criteria for Nuclear Power Plants,” in 10 CFR Part 50 for minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which CPs have previously been issued by the Commission.

The NRC regulations in 10 CFR Part 50, Appendix A, include requirements for the design, qualification, and testing of the CRD system in nuclear power plants. Regarding the CRD system in CRN-1, these requirements include, for example, the following GDC:

- GDC 1 – Quality standards and records
- GDC 2 – Design bases for protection against natural phenomena
- GDC 14 – Reactor coolant pressure boundary
- GDC 26 – Reactivity control system redundancy and capability
- GDC 27 – Combined reactivity control systems capability
- GDC 29 – Protection against anticipated operational occurrences

The NRC regulations in 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” specify QA requirements for the design, fabrication, construction, testing, and records control of certain SSCs in a nuclear power plant.

The NRC regulations in 10 CFR 50.55a, “Codes and standards,” incorporate by reference with conditions the ASME *Boiler and Pressure Vessel Code* (BPVC), Section III, “Rules for Construction of Nuclear Facility Components,” and Section XI, “Rules for Inspection of Nuclear Power Plant Components,” and ASME *Operation and Maintenance of Nuclear Power Plants*, Division 1, OM Code: Section IST (OM Code) for the design, inspection, and testing of applicable nuclear power plant components. These regulatory requirements apply to certain components of the CRD system for CRN-1.

In RG 1.70 (Revision 3 NRC 1978-TN12879), “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. Section 3.9.4, “Control Rod Drive Systems,” of

[2] From the early days of nuclear reactor operations, SCRAM is the acronym for “safety control rod axe man.”

RG 1.70 indicates that an applicant should provide (1) descriptive information of the CRD system; (2) the applicable design specifications; (3) applicable design loads, stress limits, and allowable deformations; and (4) plans for a performance assurance program.

In its review, the NRC staff evaluated whether the applicant complied with the NRC regulatory requirements in 10 CFR 50.34(a) by satisfying the NRC guidance in RG 1.70 as applicable for a new CPA with respect to the CRD system for CRN-1. The NRC staff technical evaluation of the CRN-1 PSAR Section 3.9.4 with respect to the applicable regulatory requirements and guidance is described in the following section of this NRC safety evaluation.

3.9.4.3 Technical Evaluation

The NRC regulations in 10 CFR 50.34, "Contents of applications; technical information," in paragraph (a), *Preliminary safety analysis report*, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. In RG 1.70 (NRC 1978-TN12879), the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. Section 3.9.4 of RG 1.70 provides guidance for CP applicants to meet the 10 CFR 50.34(a) requirements with respect to the CRD system for a proposed nuclear power plant. The NRC staff reviewed PSAR Section 3.9.4 for compliance with the NRC regulations in 10 CFR 50.34(a), and consistency with the NRC guidance in RG 1.70, as applicable to a new CPA, for the content of the PSAR with respect to the CRD system for the CRN-1. This SE section describes the NRC staff review of PSAR Section 3.9.4 with respect to the information required by 10 CFR 50.34(a) for the contents of the PSAR as part of a CPA with applicable guidance in RG 1.70 for the CRD system in CRN-1.

The applicant describes the CRD system for CRN-1 in detail in PSAR Section 4.6. As noted in the PSAR, the CRD system is an SC1 system, and portions of the CRD system are part of the RCPB with the system designed, fabricated, and tested to quality standards commensurate with the safety category functions that are performed. The CRD system includes the FMCRD mechanisms, hydraulic control unit (HCU) assemblies, and CRD hydraulic subsystem. The CRD system extends inside the reactor pressure vessel (RPV) to the coupling interface with the control rod blades.

In the PSAR, the applicant states that the CRN-1 BWRX-300 design will comply with the relevant requirements of 10 CFR Part 50, Appendix A, and 10 CFR 50.55a regulations. In PSAR Section 3.9.4, the applicant specifies that the CRD system complies with the following GDC:

- GDC 1 – Quality standards and records, as it relates to the CRD system
- GDC 2 – Design bases for protection against natural phenomena, as it relates to the CRD system
- GDC 14 – Reactor coolant pressure boundary, as it relates to the CRD system, requires that the RCPB portion of the CRD system be designed, constructed, and tested for the extremely low probability of leakage or gross rupture. The design involves meeting the ASME BPV Code, Section III, Subsection NB, acceptance criteria, as incorporated by reference in 10 CFR 50.55a, utilizing load combinations in PSAR Table 3.9-3, "Load Combinations and Acceptance Criteria," including transient plant duty cycles in PSAR Table 3.9-2, "Cycles of Events," for the 60-year life period to assure minimal abnormal leakage.
- GDC 26 – Reactivity control system redundancy and capability, as it relates to the CRD system, requires that the control rods are used in the reactivity control system, preferably including a positive means for inserting the rods, and shall be capable of reliably

controlling reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits (SAFDLs) are not exceeded.

- GDC 27 – Combined reactivity control systems capability, as it relates to the CRD system, requires that the reactivity control systems be designed to reliably control reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.
- GDC 29 – Protection against anticipated operational occurrences, as it relates to the CRD system requires that the CRD system, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of AOOs. The design of the CRD system to ASME BPVC, Section III, Service Level A through D and test conditions as outlined in PSAR Table 3.9-3 assures accomplishment of safety functions in the event of an AOO, thus complying with GDC 29.

The NRC staff finds that these explanations for CRD system compliance with GDC provide sufficient information to show how the relevant requirements in 10 CFR Part 50, Appendix A are met. Thus, the NRC staff find that the applicant meets the requirements of 10 CFR 50.34(a)(3)(i) with respect to the CRD system for CRN-1.

As stated in Chapter 17, “Quality Assurance,” of this report, the NRC staff has determined that Topical Report NNP-TR-001-NP-A, “Quality Assurance Program Description for TVA New Nuclear,” Revision 2, dated September 15, 2025 (ML25258A031), as incorporated by reference in Section 1.6.1, “Topical Reports,” of the CRN-1 PSAR, complies with the requirements of Appendix B to 10 CFR Part 50, as well as the requirements of 10 CFR 50.34(a)(7), which requires that an applicant for a CP provide a description of the QA program to be applied to the design, fabrication, construction, and testing of the SSCs of the facility, including the CRD system.

In PSAR Section 3.9.4, the applicant addresses the guidance provided in RG 1.70 for the CRD system in a nuclear power plant. The applicant specifies the following regarding the CRD system:

- Regarding descriptive information for the CRD system, the applicant specifies that descriptive information of the CRD system in CRN-1 is contained in PSAR Section 4.6.
- Regarding applicable design specifications, the applicant specifies that the CRD system for CRN-1 will be designed to meet the functional design criteria outlined in Section 4.6 for the electro-hydraulic FMCRD mechanisms, hydraulic control unit, and CRD hydraulic subsystem. The subcomponents of the FMCRD forming part of the RCPB will be designed according to ASME BPVC, Section III, Subsection NB, as incorporated by reference in 10 CFR 50.55a. The quality group classification of the components of the CRD system is listed in PSAR Table 3A-1, “Preliminary BWRX-300 Component Classification List,” and are designed to the codes and standards in accordance with their individual quality groups. Design and qualification of the CRD system components for CRN-1 are discussed in PSAR Section 3.9.1.1, Section 3.9.1.4.1, Section 3.9.1.4.2, Section 3.9.3.6, and Section 3.9.3.7.
- Regarding applicable design loads, stress limits, and allowable deformations, the applicant specifies that the ASME BPVC, Section III, Subsection NB components of the CRD system will be evaluated analytically, with the design loading conditions, and stress criteria specified in PSAR Table 3.9-3.

- Regarding plans for a performance assurance program, the applicant specifies that the assurance of the CRDM RCPB components to perform throughout the 60-year design life of the system will be confirmed by the ASME BPVC, Class 1 design report required by ASME BPVC, Section III, Subsection NB, as incorporated by reference in 10 CFR 50.55a. The applicant states that PSAR Section 4.6 specifies the required tests to assure CRD operability.

Based on its review, the NRC staff finds that PSAR Section 3.9.4 provides acceptable information related to the CRD system that meets the relevant parts of 10 CFR 50.55a as well as the requirements of 10 CFR 50.34(a) through the applicable guidance in RG 1.70 to support issuance of a CP for CRN-1.

3.9.4.4 Conclusion

The NRC staff has reviewed the information provided in PSAR Section 3.9.4 and, for the reasons given above, concludes that the CP applicant's description of its plans for meeting the requirements of (1) 10 CFR Part 50, Appendix A, GDC 1, and 10 CFR 50.55a with respect to designing components important to safety for the CRD system to quality standards commensurate with the importance of the safety functions to be performed, (2) GDC 2, 14, and 26 with respect to designing the CRD system to withstand the effects of earthquakes and conditions of normal operation with adequate margins to assure the system's reactivity control function and with extremely low probability of leakage or gross rupture of the RCPB, and (3) GDC 27 and 29 with respect to designing the CRD system to assure its capability of controlling reactivity and cooling the reactor core with appropriate margin in conjunction with either the emergency core cooling system or the reactor protection system, is sufficient and meets the regulatory requirements of 10 CFR 50.34(a). Therefore, the NRC staff concludes that the information provided by the applicant is sufficient to support the issuance of a CP for pursuant to the NRC regulations in 10 CFR 50.35.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Introduction

This section provides the description, design, and function of the reactor pressure vessel internals.

3.9.5.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary safety analysis report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is related to the design and function of the reactor pressure vessel internals.
- In 10 CFR Part 50, Appendix A, GDC 1, the NRC requires, in part, that reactor internals be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- In 10 CFR Part 50, Appendix A, GDC 2, the NRC requires, in part, that reactor internals be designed to withstand the effects of earthquakes without loss of capability to perform their safety function.

- In 10 CFR Part 50, Appendix A, GDC 4, the NRC requires that reactor internals be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including lost-of-coolant accidents.
- In 10 CFR Part 50, Appendix A, GDC 10, the NRC requires that reactor internals be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

SRP Section 3.9.3, "ASME Code Class 1, 2, and 3 Components and Component Support," lists acceptance criteria adequate to meet the above requirements. The following references contain acceptance criteria for this review.

- *ASME BPVC, Section III, Subsection NG, "Core Support Structures*" as it relates to the design of core support structures and internal structures.
- RG 1.29, "Seismic Design Classification for Nuclear Power Plants" as it relates to seismic classification of reactor internal components.

3.9.5.3 Technical Evaluation

PSAR, Section 3.9.5, "Reactor Pressure Vessel Internals," refers to PSAR Section 4.1.2, "Reactor Internal Components," for the description, design, and function of RVIs. PSAR Section 4.1.2 states that the major reactor internal components include core support structures, chimney, chimney head and steam separator assembly, steam dryer assembly, and associated components and system interfaces.

Section 4.1.2.2, "Core Support Structures" states that core support structures are designed, fabricated, and examined in accordance with provisions of ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG, "Core Support Structures." However, there is no information related to the design standard for reactor internal structures (other than core support structures). In Audit Question A-3.9-5, the NRC staff request that TVA provide in the PSAR the standard or requirements for the design of reactor internal structures. In response to the audit question, TVA states that the reactor internals will be designed according to ASME BPVC, Section III, Subsection NG, paragraph NG-1122 and fabricated to paragraph NG-4110(b). The NRC staff finds that constructing the reactor internal structure according to the ASME BPVC will ensure that the reactor internal structures will not affect adversely the integrity of the core support structures, and therefore acceptable.

Core support structures and internal structures are classified as seismic Category I in PSAR Table 3A-1, "Preliminary BWRX-300 Component Classification List," and therefore, the NRC staff finds that the seismic classification is consistent with RG 1.29. PSAR Table 3A-1 classifies other internal structures (e.g., chimney head and steam separator assembly, steam dryer assembly, etc.) as seismic Category II. The NRC staff notes that these internal structures are SC3 and therefore, not safety related. RG 1.29 Position C.1.b indicates that reactor vessel internals (RVIs) should be classified as seismic Category I; however, the NRC staff recognizes that RVIs consist of core support structures and internal structures. While core support structures are safety related and should be classified as seismic Category I, some internal structures are not safety related, and a seismic classification less than seismic Category I is reasonable provided they are designed and constructed such that a safe shutdown earthquake (SSE) would not cause failure that will affect the safety function of safety-related RVIs (e.g., seismic Category I SSCs). The seismic Category II designation ensures that seismic interaction with seismic Category I SSCs is considered. Therefore, the NRC staff finds that the seismic Category II classification for these non-safety-related internal structures is acceptable.

The NRC staff finds that the classification of the reactor pressure vessel internals is consistent with RG 1.29 and the design of the reactor pressure vessel internals in accordance with ASME BPVC, Section III, Subsection NG and SRP 3.9.5, and therefore meet the relevant requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, 4, and 10. As such, the design bases for methods of design for reactor pressure vessel internals for the preliminary design have been explained sufficiently under 10 CFR 50.34(a)(3)(ii).

3.9.5.4 Conclusion

The staff has reviewed the available information provided in the PSAR, and, for the reasons given above, concludes that the description of the plan for the design of reactor internals is acceptable for CP issuance and meets the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, 4, and 10; and is based on designing the reactor internals to quality standards commensurate with the importance of the safety functions performed, by designing components important to safety to withstand the effects of earthquakes and of normal operation, maintenance, testing, and postulated accidents (including loss-of-coolant-accident [LOCAs]) with sufficient margin to maintain their capability to perform safety functions, and by designing reactor internals with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.9.6 Functional Design, Qualification, and In-Service Testing Programs for Pumps, Valves, and Dynamic Restraints

3.9.6.1 Introduction

PSAR Section 3.9.6 describes the regulatory requirements for the functional design, qualification, and in-service testing (IST) programs for pumps, valves, and dynamic restraints (snubbers) that will perform safety functions in the CRN-1. PSAR Section 3.9.6 references other PSAR sections as applicable that specify provisions for the functional design and qualification of components in CRN-1. PSAR Section 3.9.6 describes IST activities to be performed for the components within the scope of the IST Program at CRN-1.

3.9.6.2 Regulatory Evaluation

The NRC regulations in 10 CFR 50.34 (TN249), "Contents of Applications; Technical Information," in paragraph (a), *Preliminary safety analysis report*, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. Paragraph (2) in 10 CFR 50.34(a) requires that the PSAR provide a summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. Paragraph (3) in 10 CFR 50.34(a) requires in item (i) that the PSAR provide a preliminary design of the facility including the principal design criteria for the facility with reference to 10 CFR Part 50, Appendix A, for minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission.

The NRC regulations in 10 CFR Part 50, Appendix A, include requirements for the functional design, qualification, and testing of pumps, valves, and dynamic restraints in nuclear power plants. With respect to the CRN-1 CPA, these requirements include, for example, the following GDC:

- GDC 1—Quality standards and records
- GDC 2—Design bases for protection against natural phenomena
- GDC 4—Environmental and dynamic design bases
- GDC 14—Reactor coolant pressure boundary
- GDC 15—Reactor coolant system design
- GDC 37—Testing of emergency core cooling system
- GDC 40—Testing of containment heat removal system
- GDC 43—Testing of containment atmosphere cleanup systems
- GDC 46—Testing of cooling water system
- GDC 54—Piping systems penetrating containment

The NRC regulations in 10 CFR Part 50, Appendix B, specify quality assurance (QA) requirements for the design, fabrication, construction, testing, and records control of applicable pumps, valves, and dynamic restraints in a nuclear power plant.

The NRC regulations in 10 CFR 50.55a, “Codes and Standards,” incorporate by reference the ASME OM Code, with conditions, for implementation of an IST Program to assess the operational readiness of applicable pumps, valves, and dynamic restraints in water-cooled nuclear power plants.

For new nuclear power plants (such as CRN-1), NRC regulations in 10 CFR 50.55a supplement ASME OM Code provisions with the following condition in 10 CFR 50.55a(b)(3):

> (iii) *OM condition: New reactors.* In addition to complying with the provisions in the ASME OM Code with the conditions specified in paragraph (b)(3) of this section, holders of operating licenses for nuclear power reactors that received construction permits under this part on or after August 17, 2018, and holders of combined licenses issued under 10 CFR part 52, whose initial fuel loading occurs on or after August 17, 2018, must also comply with the following conditions, as applicable:
>
> > (A) *Power-operated valves.* Licensees must periodically verify the capability of power-operated valves to perform their design-basis safety functions.
> > (B)-(C) [Reserved]
> > (D) *High risk non-safety systems.* Licensees must assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the Regulatory Treatment of Non-Safety Systems for applicable reactor designs.

In RG 1.70 (Revision 3, November 1978), the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. Section 3.9.6, “Inservice Testing of Pumps and Valves," in RG 1.70 provides guidance for the information consistent with the ASME codes at that time to be provided by an applicant for a CP or OL regarding the IST Program to be developed for the nuclear power plant. Since issuance of RG 1.70, ASME has modified its codes to address the IST requirements for dynamic restraints (snubbers) as part of the ASME OM Code rather than ASME BPV Code, Section XI. Further, ASME has revised the scope of the ASME OM Code to apply to safety-related pumps, valves, and dynamic restraints for the IST Program at nuclear power plants.

With the update of the ASME OM Code, the guidance in RG 1.70 for information to be provided in the PSAR for a new CPA at this time is summarized as follows. A test program should be provided that includes baseline preservice testing and a periodic IST program to ensure that all ASME OM Code pumps provided with an emergency power source and all ASME OM Code valves will be in a state of operational readiness to perform their safety function throughout the life of the plant. Descriptive information in the PSAR should cover the IST program of ASME OM Code pumps provided with an emergency power source. Reference tests for speed, pressure, flow rate, vibration, lubrication, and bearing temperature at normal pump operating conditions should be presented. Methods for measuring the reference values and in-service values for the pump parameters should be presented. The pump test plan and schedule should be provided and included in the technical specifications. Descriptive information in the PSAR should cover the IST program of ASME OM Code valves. The valve test program should include preservice tests, valve replacement, valve repair and maintenance, indication of valve position, and in-service tests for applicable valve categories. The valve test procedure and schedule should be provided and included in the technical specifications. The NRC regulations in 10 CFR 50.55a(f) require licensees of nuclear power plants to periodically update its IST program to meet the requirements of future revisions of the ASME OM Code. An applicant or licensee is allowed to submit requests for relief from or alternatives to the ASME OM Code requirements on a plant-specific basis with appropriate justification.

Revision 4 to RG 1.100 (NRC 2020-TN13088), "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," accepts the 2017 Edition of the ASME QME-1 Standard with regulatory positions, for the qualification of pumps, valves, and dynamic restraints to perform their safety functions.

In its review, the NRC staff evaluated whether the applicant complied with the NRC regulatory requirements in 10 CFR 50.34(a) and satisfied the NRC guidance in RG 1.70 as applicable for a new CPA with respect to the functional design, qualification, and IST program for pumps, valves, and dynamic restraints for CRN-1. Compliance by the applicant with other RGs, such as RG 1.100, will be addressed as part of the OL application review process. The NRC staff technical evaluation of PSAR Section 3.9.6 with respect to the applicable regulatory requirements and guidance is described in the following section of this NRC safety evaluation.

3.9.6.3 Technical Evaluation

The NRC regulations in 10 CFR 50.34, "Contents of Applications; Technical Information," in paragraph (a), *Preliminary safety analysis report*, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. In RG 1.70 (NRC 1978-TN12879), the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. Section 3.9.6 in RG 1.70 provides guidance for CP applicants to meet the 10 CFR 50.34(a) requirements with respect to an IST program for a proposed nuclear power plant. The NRC staff reviewed PSAR Section 3.9.6 for compliance with the NRC regulations in 10 CFR 50.34(a), and consistency with the NRC guidance in RG 1.70, as applicable to a new CPA, for the content of the PSAR with respect to an IST Program for CRN-1. In addition to reviewing the PSAR, the staff conducted an audit of information provided in the applicant's electronic reading room and interacted with applicant personnel regarding functional design, qualification, and IST program for pumps, valves, and dynamic restraints in CRN-1. As described in this SE section, the NRC staff reviewed PSAR Section 3.9.6 with respect to the information required by 10 CFR 50.34(a) for the contents of the PSAR as part of a CPA with applicable guidance in RG 1.70 for the functional design, qualification, and IST program for pumps, valves, and dynamic restraints in CRN-1.

PSAR Section 3.9.6 indicates that Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," and Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," specify the methods for qualification of mechanical equipment in CRN-1. The qualification involves both determining component functionality while maintaining structural integrity under seismic, dynamic, and environmental conditions. PSAR Sections 3.10 and 3.11 specify that American Society of Mechanical Engineers (ASME) Standard QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," will be applied for the qualification of mechanical equipment in CRN-1. PSAR Table 1.9-20, "Conformance with Regulatory Guides," references the application of Revision 4 to RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," which accepts the 2017 Edition of the ASME QME-1 Standard with regulatory positions. Table 1.9-20 references the 2023 Edition of the ASME QME-1 Standard, which has not been accepted in RG 1.100 at this time. During its review, the NRC staff requested the applicant to describe its basis for the use of the 2023 Edition of the ASME QME-1 Standard for the qualification program. In Revision 1 to the PSAR, the applicant revised PSAR Table 1.9-20 to clarify the use of RG 1.100 and the applicable editions of the ASME QME-1 standard.

PSAR Section 3.9.6 specifies compliance of the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints for CRN-1 with the applicable GDC in 10 CFR Part 50, Appendix A. In particular, PSAR Section 3.1, "Compliance with U.S. Nuclear Regulatory Commission General Design Criteria," addresses compliance with the plant-wide applicability of GDC 1 (Quality standards and records), GDC 2 (Design bases for protection against natural phenomena), and GDC 4 (Environmental and dynamic design bases).

PSAR Section 3.9.6 specifies compliance with GDC 14 (Reactor coolant pressure boundary) as it relates to designing, fabricating, erecting, and testing pumps, valves, and dynamic restraints that form the RCPB so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture designed in accordance with PSAR Section 3.9.1, Section 3.9.2, Section 3.9.3, and Section 3.10.

PSAR Section 3.9.6 specifies compliance with GDC 15 (Reactor coolant system design) as it relates to pumps, valves, and dynamic restraints in the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOO and design basis accident (DBA) event sequences designed in accordance with PSAR Section 3.9.1, Section 3.9.2, Section 3.9.3, and Section 3.10.

PSAR Section 3.9.6 specifies compliance with GDC 37 (Testing of emergency core cooling system) as it relates to designing the emergency core cooling system to permit periodic pressure and functional testing to ensure the leak tight integrity and operability and performance of its active components. The emergency core cooling system design incorporates adequate space and connections to allow for periodic pressure and functional testing.

PSAR Section 3.9.6 specifies compliance with GDC 40 (Testing of containment heat removal system) as it relates to designing periodic pressure and functional testing of the passive containment cooling system to ensure the leak tight integrity and operability and performance of its active components. The passive containment cooling system design incorporates adequate space and connections to allow for periodic pressure and functional testing.

PSAR Section 3.9.6 specifies compliance with GDC 43 (Testing of containment atmosphere cleanup systems) as it relates to designing the containment atmospheric cleanup systems to permit periodic pressure and functional testing to ensure the leak tight integrity and the operability and performance of the active components. The BWRX-300 containment atmosphere system is an inerted system design that incorporates adequate space and connections to allow for periodic pressure and functional testing as described in PSAR Section 9.3.6, “Containment Inerting System.”

PSAR Section 3.9.6 specifies compliance with GDC 46 (Testing of cooling water system) as it relates to designing the cooling water system to allow periodic pressure and functional testing to ensure the leak tight integrity and operability and performance of the active components. The cooling water system design incorporates adequate space and connections to allow for periodic pressure and functional testing.

PSAR Section 3.9.6 specifies compliance with GDC 54 (Piping systems penetrating containment) as it relates to designing piping systems penetrating containment with the capability to test periodically the operability of the isolation valves and associated apparatus and determine valve leakage acceptability. The piping system containment penetration design for CRN-1 incorporates adequate space and connections to allow for periodic pressure and functional testing.

Appendix B to 10 CFR Part 50 requires QA activities for the design, fabrication, construction, testing, and records control of applicable pumps, valves, and dynamic restraints for CRN-1. The applicant indicated that compliance with 10 CFR Part 50, Appendix B, will occur as part of FSAR phase during design development.

The CRN-1 IST Program will include periodic tests and inspections that demonstrate the operational readiness of specific pumps, valves, and dynamic restraints (snubbers) and their capability to perform their safety functions. PSAR Table 3.9-4, “Preliminary Active Valve List,” provides a preliminary list of active valves for CRN-1. The IST Program will be based on the requirements of ASME OM Code as incorporated by reference in 10 CFR 50.55a. In addition to the ASME OM Code, the applicant will be responsible for implementation of the specific conditions in 10 CFR 50.55a that supplement the ASME OM Code requirements for CRN-1.

The BWRX-300 design does not require the use of pumps to perform a specific function in shutting down the reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident. Therefore, there will be no pumps included in the IST Program for CRN-1.

PSAR Section 3.9.6 provides a summary of the current status of the IST Program for valves to support the CPA for CRN-1. PSAR Section 3.9.6 specifies that certain ASME BPV Code Class 1, 2, and 3 valves and pressure relief devices are subject to in service testing (IST) in accordance with the ASME OM Code as incorporated by reference in 10 CFR 50.55. The NRC staff notes that the scope of the ASME OM Code extends beyond ASME BPV Code Class 1, 2 and 3 components per ASME OM Code, Subsection ISTA, Paragraph ISTA-1100, “Scope.” During its review, the NRC staff requested that the applicant clarify the scope of the IST Program for CRN-1 consistent with the requirements of the ASME OM Code as incorporated by reference in 10 CFR 50.55a. In Revision 1 to the PSAR, the applicant clarified that the IST Program for valves will comply with the ASME OM Code, Subsection ISTA, Paragraph ISTA-1100, “Scope.” [Confirmatory item]

PSAR Section 3.9.6 specifies that valves subject to IST include valves installed to perform a specific function in shutting down the reactor to a safe shutdown condition, maintaining a safe shutdown condition, or mitigating the consequences of an accident. In addition, pressure relief devices used for protecting systems or portions of systems that perform a function in shutting down the reactor to a safe shutdown condition, maintaining a safe shutdown condition, or mitigating the consequences of an accident, are subject to IST.

The NRC staff noted that Reference 3.9-11 in the PSAR does not specify an edition of the ASME OM Code for the IST Program. The NRC regulations in 10 CFR 50.55a(f)(4)(i) require the initial IST Program for a new nuclear power plant to comply with the latest edition of the ASME OM Code (or accepted Code Cases) incorporated by reference in 10 CFR 50.55a no more than 18 months before the date of OL issuance. The NRC staff will confirm the specification of the appropriate edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a during the NRC review of the OL application for CRN-1.

PSAR Section 3.9.6 provides a summary of the IST requirements in Subsection ISTC, “Inservice Testing of Valves in Water-Cooled Reactor Nuclear Power Plants,” in the ASME OM Code for the categorization and testing of valves at CRN-1. Appendix I, “Inservice Testing of Pressure Relief Devices in Water-Cooled Reactor Nuclear Power Plants,” to the ASME OM Code includes IST requirements for pressure relief devices. Appendix II, “Check Valve Condition-Monitoring Program,” to the ASME OM Code allows a condition monitoring program for check valves. Appendix III, “Preservice and Inservice Testing of Active Electric Motor-Operated Valve Assemblies in Water-Cooled Reactor Nuclear Power Plants,” to the ASME OM Code includes requirements for diagnostic testing of motor-operated valves. Appendix IV, “Preservice and Inservice Testing of Active Pneumatically Operated Valve Assemblies in Nuclear Reactor Power Plants,” to the ASME OM Code includes requirements for diagnostic testing of air-operated valves.

PSAR 3.9.6 notes that the NRC allows ASME OM Code Cases to be applied in accordance with the provisions in RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” as incorporated by reference in 10 CFR 50.55a(a)(3)(iii), as an accepted alternative to specific ASME OM Code requirements. PSAR Section 3.9.6 indicates that the IST Program does not currently invoke any ASME OM Code Cases for CRN-1. ASME OM Code Cases accepted in RG 1.192 may be implemented by an applicant or licensee without prior approval by the NRC staff.

The applicant may submit a request for relief from an ASME OM Code requirement per 10 CFR 50.55a(f)(5), or an alternative to an ASME OM Code requirement per 10 CFR 50.55a(z). The NRC staff has not received any such requests at the time of preparation of this SE for CRN-1.

PSAR Section 3.9.6 states that no snubbers are currently included in the BWRX-300 design. However, PSAR Section 3.9.6 notes that this could change as the design matures. In the event that snubbers are included in the design at a future time, PSAR Section 3.9.6 provides a summary of the IST requirements for snubbers specified in the ASME OM Code as incorporated by reference in 10 CFR 50.55a. The NRC staff considers the summary of the IST Program requirements for snubbers in the ASME OM Code to be reasonable to support the issuance of a CP for CRN-1.

3.9.6.4 Conclusion

The NRC staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the applicant's description of the functional design, qualification, and IST programs for pumps, valves, and snubbers is sufficient and meets the regulatory requirements of 10 CFR 50.34(a) and other requirements identified in this section. Upon submittal of the OL application, the NRC staff will evaluate specific conformance with the NRC regulatory requirements for the functional design, qualification, and IST programs for pumps, valves, and snubbers for OL issuance for CRN-1. Therefore, the NRC staff concludes that the information provided by the applicant with respect to the functional design, qualification, and IST programs for pumps, valves, and snubbers is acceptable to support the issuance of a CP pursuant to the NRC regulations in 10 CFR 50.35.

3.9.7 Risk-Informed In-Service Testing

An applicant for a CP may propose a detailed risk-informed IST program in lieu of the IST requirements in the ASME OM Code as incorporated by reference in 10 CFR 50.55a. The CP applicant for CRN-1 has not proposed a risk-informed IST program. PSAR Section 3.9.7 states that this section will be addressed in the FSAR. Therefore, this SE section is not applicable to the CRN-1 CPA.

3.9.8 Risk-Informed In-Service Inspection of Piping

The CP applicant for CRN-1 has not proposed a risk-informed in-service inspection of piping program. PSAR Section 3.9.8 states that this section will be addressed in the FSAR. Therefore, this SE section is not applicable to the CRN-1 CPA.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

3.10.1 Introduction

The section addresses the methods of tests and analyses employed to ensure the functionality of mechanical and electrical equipment under seismic and other accident-related dynamic loadings.

3.10.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), "Preliminary Safety Analysis Report," sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is related to the methods of tests and analyses employed to ensure the functionality of mechanical and electrical equipment under the full range of normal and accident loadings, including seismic.

- In 10 CFR Part 50, Appendix A, GDC 1, the NRC requires, in part, that equipment be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- In 10 CFR Part 50, Appendix A, GDC 2 and Appendix S to 10 CFR Part 50, the NRC requires, in part, that equipment be designed to withstand the effects of earthquakes without loss of capability to perform their safety function.
- In 10 CFR Part 50, Appendix A, GDC 4, the NRC requires that equipment be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including lost-of-coolant accidents.
- In 10 CFR Part 50, Appendix A, GDC 14, the NRC requires that equipment associated with the reactor coolant pressure boundary (RCPB) be qualified so that there is an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- In 10 CFR Part 50, Appendix A, GDC 30, the NRC requires that equipment be qualified to appropriate quality standards commensurate with the importance of the safety functions to be performed.
- In 10 CFR Part 50, Appendix B, Criterion III, "Design Control,", the NRC requires, in part, verifying and checking the adequacy of design, such as by the performance of a suitable test program.

SRP Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," lists acceptance criteria adequate to meet the above requirements.

RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," provides relevant guidance for this review.

IEC/IEEE 60980-344, "Nuclear Facilities – Equipment Important to Safety – Seismic Qualification," provides methods for establishing seismic qualification procedures that will yield quantitative data to demonstrate that the equipment can meet is performance requirements.

ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," provides the requirements and guidelines for the qualification of active mechanical equipment whose function is required to ensure the safe operation or safe shutdown of a nuclear facility.

3.10.3 Technical Evaluation

This section of the report evaluates the seismic qualification of active mechanical (e.g., pumps and valves) and electrical equipment including instrumentation and control [I&C] components) that are designed to withstand the effects of earthquakes.

3.10.3.1 Seismic and Dynamic Qualification Criteria

PSAR Section 3.10.1.1, "Qualification Standards" states that seismic qualification of mechanical and electrical equipment and associated supports are considered for testing, analysis, or a combination of testing and analysis in accordance with International Electrotechnical Commission (IEC)/IEEE 60980-344, "Nuclear Facilities – Equipment Important to Safety – Seismic Qualification," Edition 1.0, 2020-10. The NRC staff notes that IEC/IEEE 60980-344 supersedes IEEE Std 344-2013, ""IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations." RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 4 endorses IEEE Std 344-2013 with exceptions. IEC develops the dual organization standard IEC/IEEE 60980-344 that adapts the International Atomic Energy Agency (IAEA) terminology and supersedes IEEE Std 344-2013; however, the technical provisions for seismic qualification of equipment remains consistent between IEC/IEEE 60980-344 and IEEE Std 344-2013; therefore, the NRC staff finds that the use of IEC/IEEE 60980-344 acceptable.

PSAR Section 3.10.1.1 also states that the seismic qualification of active mechanical equipment is performed considering the methods and requirements specified in ASME QME-1-2023, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities." The NRC staff notes that RG 1.100, Revision 4 endorses QME-1-2017 with exceptions. The seismic qualification provisions in QME-1-2017 are consistent with those in QME-1-2023. More specifically, QME-1-2023 generally endorses the guidelines in IEEE Std 344-2013 for seismic qualification of active equipment except as modified by ASME QME-1-2023. Therefore, the NRC staff finds that the use of QME-1-2023 for seismic qualification of active mechanical equipment acceptable.

PSAR Section 3.10.1.1.1, "Qualification by Actual Seismic Experience," states that in the design of the BWRX-300, there are no known experience-based qualification that are considered appropriate for qualification by actual seismic experience. The NRC staff finds that acceptable because as indicated in RG 1.100, the use of experience-based qualification has been used only to verify the seismic adequacy of safety-related equipment in the plants docketed before 1972.

PSAR Section 3.10.1.1.2, "Qualification by Similarity," states that qualification by similarity for Seismic Category I and II equipment is based on operating experience of similar equipment or to qualify multiple similar pieces of equipment by testing and/or analyzing only one of the pieces of equipment.

The NRC staff notes that IEC/IEEE 60980-344, subclause 7.2,"Seismic qualification methods" states that the similarity method of qualification relies on comparison to previous testing on similar equipment. Qualification test and analytical data are extrapolated from a reference equipment based on similarity to excitation (seismic environment), physical system, and functionality of the equipment. PSAR Section 3.10.1.1.2 expands the scope of qualification by similarity from based on testing and analysis to also based on operating experience. In Audit Question A-3.10-2, the NRC staff requests that TVA clarify the intent of the use of operating experience in the seismic qualification of equipment. In response to the audit question, TVA states that there are no known experience-based qualifications that are considered appropriate for qualification by similarity and proposed to revise the PSAR statement. The NRC staff finds that the response and the proposed PSAR change are acceptable because experience-based qualification is not considered in seismic qualification of equipment and the approach is consistent with RG 1.100.

PSAR Section 3.10.2, "Methods and Procedures for Qualifying Mechanical and Electrical Equipment," discusses the seismic qualification of equipment by testing, analysis, or a combination of testing and analysis.

3.10.3.1.1 Qualification by Testing

PSAR Section 3.10.2.2, “Qualification by Testing,” states that seismic qualification of mechanical and electrical equipment including I&C by testing is performed in accordance with the provisions of IEC/IEEE 60980-344. PSAR Section 3.10.2.2.1, “Interface Requirements” and Section 3.10.2.2.4, “Mounting of Test Specimen” provides the interface requirements for mounting of test specimen for testing. For interfaces that cannot be simulated on the test table, accelerations and frequency content at locations of interfaces with interconnecting cables, bus ducts, and conduits are specified as interface criteria. The NRC staff finds that the testing methodology including the interface requirement is consistent with the provisions in IEC/IEEE 60980-344, Clause 9.1.2, “Mounting,” and therefore, acceptable.

PSAR Section 3.10.2.2.2, “Test Methods,” states that the preferred method for seismic testing is to use triaxial, multi-frequency testing. However, if justified, biaxial and single-axis testing is acceptable. Section 3.10.2.2.2 also describes the test steps for biaxial tests. The NRC staff notes that IEC/IEEE 60980-344, Clause 9.6.6, “Multi-axis tests,” permits single-axis, biaxial, and triaxial tests for seismic qualification. Additionally, NRC staff finds that the provisions for biaxial tests are consistent with those in IEC/IEEE 60980-344, and therefore, acceptable.

PSAR Section 3.10.2.2.5, “Aging and Vibration Conditioning,” states that if equipment is subjected to vibrational loads throughout its lifetime in its in-service mounted condition, then vibration aging to its end-of-life condition is performed prior to seismic qualification. The NRC staff finds that consideration of vibration ageing due to non-seismic vibration is consistent with IEC/IEEE 60980-344, Clause 8.6.2, “Ageing from Non-Seismic Conditions,” and therefore, acceptable.

3.10.3.1.2 Qualification by Analysis

PSAR Section 3.10.2.3, “Qualification by Analysis,” states that qualification by analysis without testing may be acceptable on equipment that is only required to maintain its structural integrity to perform its safety function. Dynamic analysis or equivalent static analysis may be used to qualify the equipment, depending on the rigidity or flexibility of the equipment. If the fundamental frequency of the equipment is above the input excitation frequency (i.e., cutoff frequency of required response spectra), the equipment is considered rigid. A static coefficient analysis may be used for rigid equipment and seismic loads are determined statistically by multiplying the actual distributed weight of the equipment by a static coefficient equal to 1.5 times of the peak value of the required response spectra. The NRC staff finds that the seismic qualification analysis methods including the use of the 1.5 static coefficient are consistent with IEC/IEEE 60980-344, Clause 11.2, “Seismic analysis methods,” and therefore, acceptable.

3.10.3.1.3 Qualification by Combined Testing and Analysis

PSAR Section 3.10.2.4, “Qualification by Combined Testing and Analysis,” states that qualification by combined testing and analysis is used as a method for qualification for complex or large equipment where it is not practical to test the entire assembly or it is too large to be tested at once. The method can include using test data for validating an analytical model, obtaining test data such as modal frequency and damping as input for dynamic analysis, or performing an evaluation of multiple site configurations based on test results of worst-case configuration(s). The NRC staff finds that the qualification method by combined testing and analysis is consistent with IEC/IEEE 60980-344, Clause 12, “Combined analysis and testing, and therefore, acceptable.

3.10.3.2 Analysis or Testing of Supports

PSAR Section 3.10.3, “Methods and Procedures of Analysis or Testing of Supports for Mechanical, Electrical Equipment and Instrumentation,” states that methods and procedures of analysis and testing of supports for mechanical and electrical equipment and instrumentation are in accordance with IEC/IEEE 60980-344 and ASME QME-1 including the provisions for line-mounted (piping and duct systems) equipment. The NRC staff notes that IEC/IEEE 60980-344 and ASME QME-1 contain guidance on seismic analysis and testing of electrical and active mechanical equipment, and therefore, acceptable.

As discussed above, the NRC staff finds that the seismic qualification of mechanical and electrical equipment is consistent with the guidance in IEC/IEEE 60980-344, ASME QME-1, RG 1.100, and SRP 3.10, and therefore meet the relevant requirements of 10 CFR Part 50 (TN249), Appendix A; GDC 2, 4, 14, and 30; 10 CFR Part 50, Appendix B, Criteria XI; and 10 CFR Part 50 Appendix S. As such, the design bases for methods of seismic and dynamic qualification of mechanical and electrical equipment for the preliminary design have been explained sufficiently under 10 CFR 50.34(a)(3)(ii).

3.10.4 Conclusion

The staff has reviewed the available information provided in the PSAR, and, for the reasons given above, concludes that the description of the seismic qualification program which will be implemented for mechanical and electrical equipment, including I&C components, meets the requirements and for satisfying the applicable requirements of 10 CFR Part 50, Appendix A; GDC 2, 4, 14, and 30; 10 CFR Part 50, Appendix B, Criteria XI; and 10 CFR Part 50 Appendix S as they relate to qualification of equipment. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 50.34(a)(3)(ii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

Environmental Qualification (EQ) includes generation and maintenance of information to ensure SSCs can perform their intended design functions and remain fit for their purpose in the conditions under which they are expected to perform.

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Introduction

PSAR Section 3.11.1, “Equipment Identification and Environmental Conditions,” specifies that the CRN-1 EQ program generates and maintains a list of SC1 equipment located in harsh and mild environments. The environmental conditions consider normal, AOO, accident, and post-accident conditions, as applicable. Equipment located below the maximum flood level considers the effects of submergence and is qualified for flooding if it is required to function in this condition. PSAR Section 3.11.1 describes the environmental conditions that are applied in the qualification of electrical and mechanical equipment to be used in CRN-1.

3.11.1.2 Regulatory Evaluation

The NRC regulations in 10 CFR 50.34 (TN249), "Contents of Applications; Technical Information," in paragraph (a), Preliminary safety analysis report, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. Paragraph (2) in 10 CFR 50.34(a) requires that the PSAR provide a summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. Paragraph (3) in 10 CFR 50.34(a) requires in item (i) that the PSAR provide a preliminary design of the facility including the principal design criteria for the facility with reference in 10 CFR Part 50, Appendix A, for minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission.

The NRC regulations in 10 CFR Part 50, Appendix A, specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The GDC in 10 CFR Part 50, Appendix A, applicable to EQ of electrical and mechanical equipment include the following:

- GDC 1, "Quality Standards and Records," as it relates to SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.
- GDC 4, "Environmental and Dynamic Design Bases," as it relates to SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.
- GDC 23, "Protection System Failure Modes," as it relates to the protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.
- The NRC regulations in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," require, in part, that each holder of or an applicant for an OL under 10 CFR Part 50 shall establish a program for qualifying specific electric equipment. The NRC staff will evaluate compliance with this regulation as part of the OL application for CRN-1.

In RG 1.70 (Revision 3, November 1978), the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. RG 1.70, Section 3.11, "Environmental Design of Mechanical and Electrical Equipment," provides information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features and reactor protection systems are designed to ensure acceptable performance in all environments (e.g., normal, tests, and accident). RG 1.70, Section 3.11, indicates information that should be included concerning the design bases related to the capability of the mechanical, instrumentation, and electrical portions of the engineered safety features, and reactor protection system to perform their intended functions in the combined postaccident environment of temperature, pressure, humidity, chemistry, and radiation. RG 1.70, Section 3.11.1, "Equipment Identification and Environmental Conditions," specifies that all safety-related equipment and components (e.g., motors, cables, filters, pump seals, and shielding) located in the primary containment and elsewhere that are required to function during and subsequent to DBAs should be identified with their locations.

In its review, the NRC staff evaluated whether the applicant complied with the NRC regulatory requirements in 10 CFR 50.34(a) and satisfied the NRC guidance in RG 1.70 applicable to a CPA in describing the compliance with 10 CFR 50.55a with respect to EQ of electrical and mechanical equipment for CRN-1. The NRC staff technical evaluation of PSAR Section 3.11.1 with respect to the applicable regulatory requirements and guidance is described in the following section of this NRC safety evaluation.

3.11.1.3 Technical Evaluation

The NRC regulations in 10 CFR 50.34, "Contents of applications; technical information," in paragraph (a), *Preliminary safety analysis report*, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. In RG 1.70, the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. Section 3.11 in RG 1.70 provides guidance for CP applicants to meet the 10 CFR 50.34(a)(2) and (3) requirements with respect to EQ of electrical and mechanical equipment for a proposed nuclear power plant. The NRC staff reviewed PSAR Section 3.11 for compliance with the NRC regulations in 10 CFR 50.34(a), and consistency with the NRC guidance in RG 1.70, as applicable to a new CPA, for the content of the PSAR with respect to EQ of electrical and mechanical equipment for the CRN-1. In addition to reviewing the PSAR, the staff conducted an audit of information provided in the applicant's electronic reading room. As described in this SE section, the NRC staff reviewed PSAR Section 3.11.1 with respect to the information required by 10 CFR 50.34(a) for the contents of the PSAR as part of a CPA with applicable guidance in RG 1.70 for EQ of electrical and mechanical equipment in CRN-1.

PSAR Section 3.11.1 specifies that the EQ program generates and maintains a list of SC1 equipment located in harsh and mild environments. The qualification plan includes the following parameters, at a minimum and as applicable: the test and/or analysis sequence, environmental and/or seismic/dynamic or electromagnetic compatibility (EMC) requirements, test item functions, identification of industry codes and standards applicable to equipment, identification of the test equipment including description and calibration plan, and test item part numbers, quantity, mounting, and connection details.

The environmental conditions for electrical and mechanical equipment at CRN-1 will consider normal, AOO, accident, and post-accident conditions, as applicable. Equipment located below the maximum flood level considers the effects of submergence and is qualified for flooding if it is required to function in this condition. Post-accident monitoring equipment will consider the criteria for accident monitoring instrumentation and EQ guidance provided in IEC 63147:2017/IEEE Std 497-2016, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," in lieu of IEEE 497-2016 as endorsed by RG 1.97 (NRC 2019-TN12864), "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."

For CRN-1, the harsh environment qualification program will verify that the equipment is designed to be compatible and perform its safety functions during normal conditions, postulated environmental conditions, DBA, and post-accident conditions. Equipment located within harsh environment conditions is exposed to temperature, pressure, relative humidity, radiation, and chemical sprays to be addressed in the qualification program. Equipment determined to have a significant aging mechanism and located in a harsh environment account for the aging mechanism in the qualification program. Aging mechanisms to be analyzed for equipment located in a harsh environment include time-temperature degradation (thermal), cycle aging (wear), and normal radiation exposure. Analysis is performed to identify the environmental design bases for AOOs, normal, accident, and post-accident environments as applicable.

Equipment is qualified to the worst-case environmental conditions for the areas in which they are located for the duration that they are required to perform their Safety Category 1 function. The Safety Category 1 functions are either functional performance requirements or fail-safe requirements. A fail-safe Safety Category 1 function consists of not failing in a manner detrimental to plant safety, accident mitigation, or prevention of a Safety Category 1 function. The basis for the Safety Category 1 function is included in the qualification documentation. Although EQ by testing or analysis is not required for Safety Class 2 (SC2) and Safety Class 3 (SC3) components, these components are designed for their expected duty cycle and environmental conditions over the design life of the plant with consideration of maintenance and aging management. Additionally, any non-SC1 electrical component whose failure under postulated environmental conditions could prevent satisfactory accomplishment of a Safety Category 1 function is qualified to the specified environmental conditions by testing or analysis in accordance with 10 CFR 50.49(b)(2).

The environments considered for electrical and mechanical equipment in the EQ program include temperature, pressure, humidity, chemical effects, radiation, and flooding effects. The EQ program uses the recommended environmental margins per Table 1 in IEC/IEEE Standard 60780-323, "Nuclear Facilities – Electrical Equipment Important to Safety – Qualification." Aging requirements apply to SC1 equipment. For equipment located in harsh and mild environments, the effect of aging is evaluated prior to DBA testing when a significant aging mechanism exists. Equipment is reviewed in terms of design, function, materials, and environment for its specified application to identify potentially significant aging mechanisms. Equipment that could be exposed to radiation is environmentally qualified to a radiation dose that simulates the calculated radiation environment (normal and accident) that the equipment can withstand prior to completion of its required safety functions. Radiation qualification considers that equipment damage is a function of total integrated dose and can be influenced by dose rate, energy spectrum, and particle type. The radiation qualification includes doses from all potential radiation sources at the equipment location. For equipment that is required to be functional post-accident, then the radiation dose is increased beyond the dose required for qualified life to envelop post-accident conditions as well, unless it is determined to cover post-accident conditions separately.

A mild radiation environment for electronic equipment is defined as a total integrated dose less than 10 gray (Gy) (1.0×10^3 rad), and a mild radiation environment for other equipment is less than 100 Gy (1.0×10^4 rad) as defined in RG 1.89. Electronic and electrical equipment are tested with the equipment energized and performing its safety function if the required total integrated dose exceeds the mild environment level. This ensures equipment will be qualified for the worst-case radiation with DBA margin per the requirements of IEC/IEEE 60780-323.

Regarding electromagnetic interference/radio frequency interference (EMI/RFI) and voltage surges, EMC requirements apply to all levels of safety class equipment, SC1, SC2, SC3, and Non-Safety Class (SCN) and provides qualification methods and implementation guidance. EMC qualifications for the BWRX-300 design follow the requirements defined in (1) Electric Power Research Institute (EPRI) TR-102323, "Guidelines for Electromagnetic Compatibility Testing of Power Plant Equipment," or (2) Military Standards MIL-STD-461G, "Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment," or (3) IEC 62003, "Nuclear Power Plants – Instrumentation, Control, and Electrical Power Systems – Requirements for Electromagnetic Compatibility Testing." The qualification for electromagnetic interference/radio frequency interference (EMI/RFI) and voltage surges for EQ of equipment in harsh and mild environments will be by test, and will be consistent with RG 1.180 (NRC 2019-TN12943), "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems." EMC qualification and acceptance testing includes tests for susceptibility and emissions. Susceptibility and emissions requirements are applied to all SC and SCN microprocessor-based I&C and electrical equipment.

Based on its review, the NRC staff finds that PSAR Section 3.11.1 provides acceptable information related to the equipment identification and environmental conditions for EQ of electrical and mechanical equipment that meets 10 CFR 50.34(a) consistent with the applicable guidance in RG 1.70 to support the CPA for CRN-1.

3.11.1.4 Conclusion

The NRC staff has reviewed the available information provided in the PSAR Section 3.11.1 and, for the reasons given above, concludes that the applicant's description of the equipment identification and environmental conditions for EQ of electrical and mechanical equipment is sufficient and meets the regulatory requirements of 10 CFR 50.34(a) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.11.2 Qualification Tests and Analyses

3.11.2.1 Introduction

PSAR Section 3.11.2, "Qualification Tests and Analyses," describes methods by testing and analyses to be applied in EQ of electrical and mechanical equipment in support of the CPA for CRN-1.

3.11.2.2 Regulatory Evaluation

The NRC regulations in 10 CFR 50.34 (TN249), "Contents of Applications; Technical Information," in paragraph (a), Preliminary Safety Analysis Report, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. Paragraph (2) in 10 CFR 50.34(a) requires that the PSAR provide a summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. Paragraph (3) in 10 CFR 50.34(a) requires in item (i) that the PSAR provide a preliminary design of the facility including The principal design criteria for the facility with reference to 10 CFR Part 50, Appendix A, for minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission.

The NRC regulations in 10 CFR Part 50, Appendix A, specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for SSCs important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

The GDC in 10 CFR Part 50, Appendix A, applicable to EQ of electrical and mechanical equipment include the following:

- GDC 1, Quality Standards and Records, as it relates to SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDC 2, Design Bases for Protection Against Natural Phenomena, as it relates to SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.
- GDC 4, Environmental and Dynamic Design Bases, as it relates to SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.
- GDC 23, Protection System Failure Modes, as it relates to the protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

The NRC regulations in 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” require, in part, that each holder of or an applicant for an OL under 10 CFR Part 50, shall establish a program for qualifying the specific electric equipment. The NRC staff will review compliance with this regulation as part of the OL application for CRN-1.

The NRC regulations in 10 CFR 50.55a incorporate by reference ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” with conditions. ASME BPVC, Section III, includes specific requirements for the construction, design, material, fabrication, examination, testing, and certification of nuclear facility components (including vessels, piping, pumps, valves, and supports) to be used in nuclear power plants.

In RG 1.70 (Revision 3; NRC 1978-TN12879), the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. RG 1.70, Section 3.11, “Environmental Design of Mechanical and Electrical Equipment,” provides information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features and reactor protection systems are designed to ensure acceptable performance in all environments (e.g., normal, tests, and accident). RG 1.70, Section 3.11 indicates information that should be included concerning the design bases related to the capability of the mechanical, instrumentation, and electrical portions of the engineered safety features, and reactor protection system to perform their intended functions in the combined postaccident environment of temperature, pressure, humidity, chemistry, and radiation. RG 1.70, Section 3.11.2, “Qualification Tests and Analyses,” indicates that a description should be provided of the qualification tests and analyses that have been or will be performed to ensure that equipment within the scope of the EQ program will perform their function in the combined temperature, pressure, humidity, chemical, and radiation environment.

In its review, the NRC staff evaluated whether the applicant complied with the NRC regulatory requirements in 10 CFR 50.34(a) and satisfied the NRC guidance in RG 1.70 applicable to a CPA in describing the compliance with 10 CFR 50.55a with respect to EQ of electrical and mechanical equipment for CRN-1. The NRC staff technical evaluation of PSAR Section 3.11.2 with respect to the applicable regulatory requirements and guidance is described in the following section of this NRC safety evaluation.

3.11.2.3 Technical Evaluation

The NRC regulations in 10 CFR 50.34, "Contents of Applications; Technical Information," in paragraph (a), *Preliminary Safety Analysis Report*, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. In RG 1.70 (NRC 1978-TN12879), the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. Section 3.11 in RG 1.70 provides guidance for CP applicants to meet the 10 CFR 50.34(a) requirements with respect to EQ of electrical and mechanical equipment for a proposed nuclear power plant. The NRC staff reviewed PSAR Section 3.11 for compliance with the NRC regulations in 10 CFR 50.34(a), and consistency with the NRC guidance in RG 1.70, as applicable to a new CPA, for the content of the PSAR with respect to EQ of electrical and mechanical equipment for CRN-1. In addition to reviewing the PSAR, the staff conducted an audit of information provided in the applicant's electronic reading room. As described in this SE section, the NRC staff reviewed PSAR Section 3.11.2.2 with respect to the information required by 10 CFR 50.34(a) for the contents of the PSAR as part of a CPA with applicable guidance in RG 1.70 for EQ of electrical and mechanical equipment in CRN-1.

Regarding qualification by testing, PSAR Section 3.11.2.1 states that type testing is the preferred method for demonstrating that equipment is environmentally qualified. A type test subjects a representative sample of equipment, including interfaces, to a series of tests, and includes simulating the effects of significant aging mechanisms during normal operation. The sample is subsequently subjected to conditions that simulate DBA harsh conditions and thereby establishes the tested configuration for installed equipment service, including mounting, orientation, interfaces, conduit sealing, and expected environments. A type test demonstrates that the equipment performs the intended Safety Category functions for the required operating time before, during, and/or following the DBA, as appropriate. Tests are performed in accordance with applicable industry standards.

Regarding qualification by analysis, PSAR Section 3.11.2.2 states that in general, analysis is used to supplement test data, and the analytical techniques and modeling assumptions are, when possible, based on a correlation of the analytical approach with testing or operating experience performed on similar equipment or structures. Seismic and dynamic qualification by analysis is described in PSAR Section 3.10.2.3. For qualification by analysis, a logical assessment, or a valid mathematical model of the equipment to be qualified is required, and the basis for the analysis includes physical laws of nature, results of test data, operating experience, and condition indicators, as applicable. Analysis of data and tests for material properties, equipment rating, and environmental tolerance are acceptable methods to be used to demonstrate qualification. PSAR Section 3.11.2.2 indicates that analysis alone is not used to demonstrate the initial qualification for electrical equipment in a harsh environment.

Regarding qualification by operating experience, PSAR Section 3.11.2.3 states that qualification by use of operating experience requires documented data to be available to confirm that the following conditions are met:

- The component providing the operating experience is identical or justifiably similar to the
- equipment to be qualified.
- The component providing the operating experience has operated under service conditions which equal or exceed in severity the service conditions and performance requirements for which the component is to be qualified are bounded by the component providing the operating experience.
- The installed component in general is removed from service and subjected to partial type testing to include the DBA environments for which the component is to be qualified.

Regarding combined qualification, PSAR Section 3.11.2.4 states that a combination of test and analysis is used when it is deemed practical to use both methods to complete the qualification. The combined qualification method can be used for qualification for larger electrical equipment where it is not practical to test the entire assembly, or it is too large to be tested at once, unless another method of qualification is justified.

For digital I&C equipment qualification in a mild environment, analysis can be used in addition to testing if there is testing of an identical item of equipment under identical conditions or under similar conditions or operating experience with a supporting analysis to show that the equipment to be qualified is acceptable. Also, I&C equipment qualification may be performed using an analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

Regarding qualification of mechanical equipment, PSAR Section 3.11.2.5.1 states that SC1 mechanical equipment, with the Safety Category 1 function of maintaining pressure integrity, and which is designed, fabricated, and qualified consistent with ASME BPV Code, Section III, Division 1, is considered qualified. Where the loading under normal service is more severe than loading under DBA, then the loading under normal service is considered in addition to the loading under DBA by test and/or analysis. The loading and capability under DBA conditions is analyzed in the qualification process to establish the suitability of materials, parts, and equipment needed for safety functions, and to verify that the design of such materials, parts, and equipment is adequate. The qualification of mechanical equipment includes, as applicable, materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms), required operating time, non-metallic subcomponents of such equipment, the environmental conditions and process parameters for which this equipment is qualified, non-metallic material capabilities, and the evaluation of environmental effects. The qualification guidance provided in the ASME QME-1 Standard is considered for qualification of SC1 valves and SC1 mechanical pipe supports. The qualification of non-metallic parts considers the qualification guidance provided in ASME QME-1, Nonmandatory Appendix QR-B, "Guide for Qualification of Nonmetallic Parts."

With respect to the qualification of electrical equipment, PSAR Section 3.11.2 states that qualification guidance will be considered for specific electrical equipment, if applicable, in the following RGs:

- RG 1.158, "Qualification of Safety-Related Vented Lead-Acid Storage Batteries for Nuclear Power Plants," which endorses IEEE 535-2022, "IEEE Standard for Qualification of Class

1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations" (NRC 2018-TN13178)

- RG 1.40, "Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants," which endorses IEEE 334-2006, "IEEE Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations" (NRC 2010-TN8123)
- RG 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants," which endorses IEEE 317-2013, "IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations" (NRC 1987-TN13179)
- RG 1.73, "Qualification Tests for Safety-Related Actuators in Nuclear Power Plants," which endorses IEEE 382-2019, "IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations and Other Nuclear Facilities" (NRC 2024-TN13180)
- RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" which endorses IEC/IEEE-60780-323 that includes IEEE 638-2013, "IEEE Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Stations" (NRC 2023-TN12897)
- RG 1.213, "Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants," considers conformance with the requirements of IEEE 649-2006, "IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations" (NRC 2009-TN13183)
- RG 1.210, "Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants," which endorses IEEE 650-2006, "IEEE Standard for Qualification of Class 1E Static Battery Chargers, Inverters, and Uninterruptible Power Supply Systems for Nuclear Power Generating Stations" (NRC 2024-TN13182)
- The NRC will evaluate the application of these RGs during the OL review process.

Regarding qualification of I&C equipment, PSAR Section 3.11.2.5.3 states that qualification guidance will be considered for specific I&C equipment, such as control boards, panels, and racks classified as SC1 components in IEEE 420-2023, "IEEE Standard for Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations." Qualification of computer-based I&C systems will be performed in accordance IEEE 7-4.3.2-2016, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations." Guidance for the EMC requirements is specified in RG 1.180.

When computer-based I&C systems environmental type testing is performed, the system being tested demonstrates that it functions and performs with safety software that has undergone verification and validation testing and is representative of the software to be installed in service. The testing demonstrates performance of all safety functions that might be affected by environmental factors under the environmental service conditions specified in the design specification. Software algorithms, which are tested during verification and validation (V&V) testing, are not tested unless their outputs exercise different hardware components that are affected by environmental conditions. The testing exercises all portions of the system that are necessary to accomplish the safety functions and those portions whose operation or failure could impair the safety functions. The testing confirms the response of digital interfaces and verifies that the design accommodates the potential effect of environmental conditions on the overall response of the system. When testing of a complete system is not practical, confirmation of the dynamic response to the most limiting environmental and operational conditions is based on type testing of the individual modules and analysis of the cumulative effects of environmental and operational stress on the entire system to demonstrate required safety performance.

For qualification of SC1 cables, PSAR Section 3.11.2.5.4 states that the qualification guidance provided in IEEE 383-2023, “IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities,” and IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” is considered. The test requirement guidance provided in IEEE 1202-2023, “IEEE Standard for Flame-Propagation Testing of Wire and Cable,” is used as a qualification program.

Seismic Category I supports (hangers) that support trays or conduits that carry SC1 circuits are designed and analyzed to demonstrate qualification in accordance with IEEE 628-2020, “IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations.” Seismic Category II supports, used for SCN raceway (conduit and cable tray) in Seismic Category I and II structures are analyzed to withstand the effects of an SSE.

SC1 connection assemblies will consider the qualification guidance provided in IEEE 572-2019, “IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations and Other Nuclear Facilities,” as endorsed by RG 1.156 (NRC 2023-TN13185), “Qualification of Connection Assemblies for Production and Utilization Facilities.”

With respect to line-mounted equipment, Clinch River PSAR Section 3.11.2.5.5 states that guidance in IEC/IEEE 60980-344, “Nuclear Facilities Equipment Important to Safety-Seismic Qualification,” identifies that special consideration is provided for line-mounted (pipe-supported) equipment regarding seismic qualification as the most critical seismic loading condition that occurs as a result of the piping or duct system. Guidance for line-mounted equipment is provided in IEEE 382-2019. Line-mounted equipment also includes active mechanical equipment subject to the provisions in the ASME QME-1 Standard including Non-mandatory Appendix QR-A, “Seismic Qualification of Active Mechanical Equipment.”

Based on its review, the NRC staff finds that Clinch River PSAR Section 3.11.2 provides acceptable information related to qualification tests and analyses for EQ of electrical and mechanical equipment that meets 10 CFR 50.34(a) consistent with the applicable guidance in RG 1.70 to support the CPA for CRN-1. In particular, the applicant meets 10 CFR 50.34(a)(2) by providing a summary description of the facility and methods by testing and analyses to be applied in EQ of electrical and mechanical equipment in support of the issuance of a CP for CRN-1. The applicant meets 10 CFR 50.34(a)(3)(i) by providing a preliminary design of the facility including the principal design criteria under 10 CFR Part 50, Appendix A, that are relevant to this section of this report, as discussed above, with a description of methods for testing and analyses to be applied in EQ of electrical and mechanical equipment in support of issuance of the CP for CRN-1.

3.11.2.4 Conclusion

The NRC staff has reviewed the available information provided in the PSAR Section 3.11.2 and, for the reasons given above, concludes that the applicant’s description of the qualification tests and analyses for EQ of electrical and mechanical equipment is sufficient and meets the regulatory requirements of 10 CFR 50.34(a) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.11.3 Conclusion

The NRC staff has reviewed the available information provided in the PSAR Section 3.11 and, for the reasons given above, concludes that the CP applicant's description of its plans for environmental design and qualification of mechanical, electrical and I&C equipment that are important to safety is applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a). Therefore, the NRC staff finds that the information provided by the CP applicant adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35. When submitted as part of an OL application, the NRC staff will evaluate the specific programs for environmental design and qualification of mechanical, electrical and I&C equipment that are important to safety for compliance with the relevant regulatory requirements of 10 CFR 50.49; 10 CFR Part 50, Appendix A, GDC 1, 2, 4, and 23; and 10 CFR Part 50, Appendix B, Quality Assurance Criteria III, XI, and XVII; with respect to systems and components being designed to withstand the effects of, and being capable of performing their safety function, in the environmental conditions associated with normal operation, maintenance, testing, and accident conditions, in support of OL issuance for CRN-1.

3.12 ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports

3.12.1 Introduction

This section covers the design and structural integrity of piping systems and supports used in Safety Class I and non- Safety Class I piping systems, the failure of which could potentially affect Safety Class I systems.

3.12.2 Regulatory Evaluation

The following NRC regulations contain the relevant requirements for this review:

- In 10 CFR 50.34(a) (TN249), “Preliminary Safety Analysis Report,” sub-paragraph (3)(ii), the NRC requires applicants to address the design bases and the relation of the design bases to the principal design criteria, which in this case is related to the design and analysis of piping systems, pipe supports, and components.
- In 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1, the NRC requires, in part, that piping systems, pipe supports, and system components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.
- In 10 CFR Part 50, Appendix A, GDC 2 and 10 CFR Part 50, Appendix S, the NRC requires, in part, that piping systems be designed to withstand the effects of earthquakes without loss of capability to perform their safety function.
- In 10 CFR Part 50, Appendix A, GDC 4, the NRC requires that piping systems be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.
- In 10 CFR Part 50, Appendix A, GDC 14, the NRC requires that the RCPB be qualified so that there is an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- In 10 CFR Part 50, Appendix A, GDC 15, the NRC requires that RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design condition of the RCPB is not exceeded during any condition of normal operation, including AOOs.

SRP Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports,” lists acceptance criteria adequate to meet the above requirements. The following reference documents provide guidance for this review:

- SRP Section 3.9.2, “Dynamic Testing and Analysis of Systems, Structures, and Components”
- SRP Section 3.7.2, “Seismic System Analysis”
- ASME BPV Code, Section III
- NUREG-1061, Volume 4, “Report of the U.S. Nuclear Regulatory Commission Piping Committee”
- RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors” (NRC 2018-TN8151)
- NUREG-1367, “Functional Capability of Piping Systems”
- RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis” (NRC 2012-TN13038)
- RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants”
- Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems”
- NUREG/CR-6909, “Effect of LWR Environments on the Fatigue Life of Reactor Materials”

3.12.3 Technical Evaluation

- The evaluation of the general design aspects for piping systems, piping components, and pipe supports is provided in the following sections.

3.12.3.1 Piping Analysis Methods

PSAR Section 3.12.1, “Piping Analysis Methods,” states that ASME Class 1 piping design conforms to the requirements of ASME BPV Code. Section III, Subsection NB, “Class 1 Components.” ASME Class 2 and 3 piping conforms to the requirements of ASME BPV Code, Section III, Subsection NCD, “Class 2 and Class 3 Components.” The containment penetration sleeves of ASME Class 2 piping are an anchor for the piping. The sleeves of the containment structure penetrations meet the requirements of ASME BPV Code, Section III, Subsection NE, “Class MC Components.” The NRC staff finds that 10 CFR 50.55a (TN249) requires the use of ASME BPV Code, Section III, Subsection NB for the design of ASME Class 1 piping and Subsection NCD for ASME Classes 2 and 3 piping. 10 CFR 50.55a also endorses the use of ASME BPV Code, Section III, Subsection NE for design of metal containment components. Therefore, the NRC staff find the piping system design methods acceptable.

PSAR Section 3.12.1.1, “Experimental Stress Analysis Methods,” states that stress intensification factors and flexibility factors for fittings and joint designs are established using experimental or analytical data. Determination of non-standard stress intensification factors and flexibility factors is in accordance with the rules of ASME BPV Code, Section III, Subsection NB, paragraph NB-3672.7. The NRC staff notes that PSAR Section 3.9.1.3, “Experimental Stress Analysis,” states that experimental stress analysis methods are used in compliance with the provisions of ASME BPV Code, Section III, “Appendices.” The NRC staff finds that using ASME BPV Code including mandatary Appendix II, “Experimental Stress Analysis and Determination of Stress intensification Factors,” is consistent with the guidance in SRP 3.12, and therefore, acceptable.

PSAR Section 3.12.1.2, “Modal Response Spectrum Method,” states that the seismic response of the piping system is determined by performing a modal analysis by either response spectrum method or time history method. The support movement of multiple-supported piping and equipment systems subject to seismic and other dynamic load excitations is considered in design verification. The computed individual maximum support displacements are imposed on the supported piping and equipment systems in the most limiting, or severe, directional combination for the piping and equipment design verification consideration. The NRC staff finds that the use of response spectrum method or time history method for seismic piping analysis is consistent with SRP 3.12 and the calculation of piping system support displacements based on the most limiting load/directional combination is conservative and is consistent with the SRP 3.9.2 guidance and therefore, acceptable.

PSAR Section 3.12.1.3, “Response Spectra Method – Independent Support Motion Method,” states that the independent support motion or multiple response spectra method is used when components in a piping system are supported by multiple support structures or multiple levels within a structure. The combination of modal responses and spatial components for systems analyzed using independent support motion is performed consistent with NUREG-1061, “Evaluation of Other Loads and Load Combinations.” Inertia response from multiple groups in the same direction are combined by absolute summation method. Modal and directional response are combined by square-root-of-the-sum-of the-square (the SRSS) method without considering closely spaced frequencies. The responses due to relative displacements at the support points are combined with the inertial responses by the SRSS method. The NRC staff finds that the combination of relative displacements and dynamic (inertia) response for the independent support motion method is consistent with SRP 3.12 and Section 2, “Staff Recommendations on Response Combinations,” of NUREG-1061, Volume 4, “Report of the U.S. Nuclear Regulatory Commission Piping Committee,” and therefore acceptable.

PSAR Section 3.12.1.4, “Time History Method,” states that the three components of design ground motion analyses conform to the acceptance criteria in RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis.” The modal and spatial combinations in the time history method also conform to RG 1.92. The NRC staff finds that the time history analysis method is consistent with the guidance in SRP 3.12 and SRP 3.7.2 and therefore is acceptable.

PSAR Section 3.12.1.5, “Inelastic Analysis Method,” states that piping system stresses are calculated on an elastic basis for each service level and evaluated in accordance with ASME BPVC, Section III, paragraph NB-3600 for ASME Class 1 piping and paragraph NCD-3600 for ASME Classes 2 and 3 piping. Inelastic analysis method is not used in BWRX-300 piping design and analysis. The NRC staff finds that the use of elastic analysis for the ASME Classes 1, 2, and 3 piping in accordance with ASME Section III requirements is consistent with 10 CFR 50.55a and SRP 3.12, and therefore, acceptable.

PSAR Section 3.12.1.6, “Non-Seismic and Seismic Category II” refers to PSAR Section 3.7.3.8 for the design of the interaction of non-seismic Category I subsystems with SSCs that are categorized as seismic Category I. PSAR Section 3.12.1.6 further states that in certain instances, seismic Category II piping is connected to seismic Category I piping at locations other than a piece of equipment which, for purposes of analysis, could be represented as an anchor. The transition points typically occur at seismic Category I valves, which may or may not be physically anchored. The dynamic piping analysis model is from pipe anchor point to anchor point, using two options. One option is to specify and design a structural anchor at the seismic Category I valve and analyze the seismic Category I subsystem. The second option is to analyze the subsystem from the anchor point in the seismic Category I subsystem through the valve to either the first anchor point in the seismic Category II subsystem, or for a distance such that there are at least two seismic restraints in each of the three orthogonal directions. The NRC staff finds that the analysis of the seismic Category II piping to ensure its failure will not affect the safety function of the seismic Category I piping meets the guidance in in SRP 3.12 and SRP 3.9.2 and therefore is acceptable. In PSAR Section 3.7.3.12, the applicant stated that the BWRX-300 design does not include seismic Category I buried pipes, conduits, or tunnels. Therefore, no staff evaluation is required (from Section 3.7.3.3.12 of this report)

3.12.3.2 Piping Modeling Techniques

PSAR Section 3.12.2.1, “Computer Codes” notes that computer codes used in the mechanical system and component analyses of the SC1 components are described in Appendix 3J and PSAR Subsection 3.9.1.2, and staff evaluated in those in Appendix J of this report and in Section 3.9.1.3 of this report. Section 3.9.1.3 of this report also evaluates the Piping Benchmark Program discussed in PSAR Section 3.12.2.3. PSAR Section 3.12.2.4, “Decoupling Criteria,” states that decoupling is a method of dividing or separating analysis models based on pipe size/moment of inertia. Branch lines with a run to branch moment of inertia ratio of 25 to 1 or greater are decoupled. The NRC finds the coupling criterion for piping analysis acceptable because it is consistent with industry practice and it ensures that the run piping or branch piping has a negligible effect on the uncertainty due to decoupling.

3.12.3.3 Piping Stress Analyses Criteria

The piping stress analysis criteria and load combinations for ASME Class 1 piping are described in PSAR Table 3.12-1, “Load Combinations and Acceptance Criteria for ASME Class 1 Piping,” and those for ASME Classes 2 and 3 piping are delineated in PSAR Table 3.12-2, “Load Combinations and Acceptance Criteria for ASME Class 2 and 3 Piping.” The NRC staff finds that the acceptance criteria and load combinations for ASME Classes 1, 2, and 3 piping systems are consistent with the respective requirements of ASME BPVC, Section III, Subsections NB and NCD as incorporated by reference in 10 CFR 50.55a and therefore are acceptable.

PSAR Section 3.12.3.4, “Damping Ratios,” states that floor response spectra used for the qualification of ASME Class 1, 2, and 3 piping for dynamic loads with Service Level B service limits use a 3 percent critical damping ratio for piping greater than or equal to 300 mm nominal pipe size (NPS), and a 2 percent critical damping ratio for piping less than 300-mm NPS. Floor response spectra used for the qualification of ASME Class 1, 2, and 3 piping for dynamic loads with Service Level C or D service limits use a 4 percent critical damping ratio for all pipe sizes. The NRC staff finds that the critical damping ratios for piping analysis including the 4 percent critical damping ratio for the SSE event are consistent with SRP 3.12 and RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” and therefore acceptable.

PSAR Section 3.12.3.7, “Fatigue Evaluation for ASME Class 1 Piping,” states that the fatigue evaluation for ASME Class 1 piping conforms to ASME BPVC, Section III, Subsection NB requirements, which includes the fatigue usage and stress limits. The cumulative usage factor is calculated in accordance with ASME BPVC, Section III, paragraph NB-3653.5, “Cumulative Damage” and RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors.” The NRC staff finds that performing fatigue analysis of ASME Class 1 piping in accordance with ASME BPVC, Section III, Subsection NB and RG 1.207 is consistent with the guidance in SRP 3.12 and therefore is acceptable.

PSAR Section 3.12.3.8, “Fatigue Evaluation of ASME Class 2 and 3 Piping,” states that the fatigue evaluation for ASME Classes 2 and 3 piping conforms to ASME BPVC, Section III, Subsection NCD requirements. Determination of non-standard stress intensification factors and flexibility factors is in accordance with the requirements of ASME BPVC, Section III, paragraph NCD-3673.2, while expansion and flexible hoses are designed in accordance with the requirements of paragraph NCD-3649. The NRC staff finds that performing fatigue analysis of ASME Classes 2 and 3 piping in accordance with ASME BPVC, Section III, Subsection NCD requirements is consistent the guidance in SRP 3.12 and therefore is acceptable.

PSAR Section 3.12.3.9, “Thermal Oscillations in Piping Connected to the Reactor Coolant System,” states that thermal fatigue of unisolable piping connected to the reactor coolant system (RCS) can occur when the connected piping is isolated by a leaking block valve, the pressure upstream from the block valve is higher than RCS pressure, and the temperature upstream is significantly cooler than RCS temperature. An unrecognized phenomenon and possibly unanalyzed condition may exist for those reactors that are subjected to these conditions. Under these conditions, thermal fatigue of the unisolable piping can result in crack initiation. If these conditions are identified, a program meeting the criteria in Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” which provides continued assurance that piping will not be subjected to unacceptable thermal stresses, will be developed. The NRC staff finds that TVA’s plan to develop a program to ensure piping will not be subjected to unacceptable thermal stresses meets the guidance in SRP 3.12 and therefore is acceptable.

PSAR Section 3.12.3.10, “Thermal Stratification,” states that transient loading due to hot and cold fluid not mixing at low flow conditions is considered in the design of the piping. The stresses and fatigue associated with thermal stratification and thermal striping are considered in the piping analyses that demonstrate compliance within ASME Class 2 and 3 code limits. The global effect of thermal stratification causes the bending of the piping system due to the temperature distribution in the element cross section; consequently, this generates extra thermal stress in the load combinations as defined in PSAR Table 3.12-2. The NRC staff finds that consideration of thermal stratification in the piping stress analysis and load combination will ensure that the potential pipe crack issue due to thermal fatigue identified in SRP 3.12, is addressed in the piping design, and therefore acceptable.

PSAR Section 3.12.3.12, “Functional Capability,” states that the functional capability of ASME Class 1, 2, and 3 piping systems essential for the safe shutdown events is ensured by meeting the piping design requirements such as:

- the conditions of Equation (9) in ASME BPVC, Section III, paragraph NB-3652 for Service Level C and D for Class 1 piping
- the conditions in Equation (9a) in ASME BPVC, Section III, paragraph NCD-3653.1(a) for Service Level C and D for Class 2 and 3 piping

- the ratio of pipe diameter to wall thickness is less than or equal to 50
- external pressure is less than or equal to internal pressure

The NRC staff finds that the design provisions will ensure functional capability of the piping systems and meet the positions in SRP 3.12 and NUREG-1367.

PSAR Section 3.12.3.14, “Operating Basis Earthquake as a Design Load,” states that BWRX-300 considers OBE loads as one-third of the SSE loads, and design load combinations that consider OBE loads for piping systems are not required, The NRC staff finds that not performing specific piping design analysis if the OBE loads are one-third or less of the SSE loads is consistent the requirements of 10 CFR 50, Appendix S and guidance in SRP 3.12.

PSAR Section 3.12.3.15, “Welded Attachments,” states that piping system and pipe supports are designed and located to minimize the necessity for welded attachments to the pipe. Where welded attachments exist, local stress evaluations are performed. The local stress due to the weld is added to the regular piping stress and will meet the ASME BPVC, Section III requirements. The NRC staff finds that including the local weld stress in the piping stress analysis will ensure the integrity of the piping systems, and therefore acceptable.

PSAR Section 3.12.3.17, “Temperature for Thermal Analyses,” states that the analysis of thermal expansion includes all thermal operating modes, environmental conditions, cold water modes, and thermal attenuation. Sufficient thermal expansion cases are established to account for various operating conditions to determine the maximum range of thermal expansion stresses. The thermal stress-free state for the piping systems is defined as a temperature of 21°C degrees Celsius (70 degrees Fahrenheit) unless a basis is provided to use a higher temperature in accordance with ASME BPVC, Section III. The NRC staff finds that analysis of the piping systems to include thermal expansion and the selection of the stress-free reference temperature is consistent with SRP 3.12 and therefore is acceptable.

PSAR Section 3.12.3.18, “Intersystem Loss-of-Coolant Accident,” states that the design includes consideration of high energy and moderate energy fluid system piping located inside and outside of containment. To the extent practical, low-pressure systems (e.g., systems and subsystems connected to the RCS that extend outside the primary containment boundary including valve stem seals, pump seals, heat exchanger tubes) are designed to withstand full RCS pressure. The NRC staff finds that designing the low-pressure system to the extent practicable to withstand the full RCS pressure will provide reasonable assurance that the failure of RCS boundary isolation will not result in rupture of the low-pressure piping system and meets the guidance in SRP 3.12.

PSAR Section 3.12.3.19, “Effects of Environment on Fatigue Design,” states that ASME Class 1 piping is evaluated for the effects of fatigue as a result of pressure and thermal transients and other cyclic events including earthquakes. In addition, PSAR Section 3.9.3.1 states that for ASME Class 1 components where analysis for cyclic operation is evaluated in accordance with ASME BPVC, Section III, paragraph NB-3222.4, the fatigue usage evaluation includes the use of environmental fatigue curves in accordance with RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors” and NUREG/CR-6909, “Effect of LWR Environments on the Fatigue Life of Reactor Materials.” The NRC staff finds that the piping fatigue analysis taking into consideration of the effects of pressure, thermal transients, earthquakes, and light-water reactor water environments is consistent with SRP 3.12 and therefore is acceptable.

3.12.3.4 Piping Support Design

PSAR Section 3.12.4.1, “Pipe Support Applicable Codes,” states that the pipe supports attached to the ASME Class 1, 2, and 3 piping meet the appropriate requirements of ASME BPVC, Section III, Subsection NF. The NRC staff finds that the applicable code for the design of ASME Class 1, 2, and 3 piping supports is acceptable and meets the guidance in SRP 3.12.

PSAR Section 3.12.4.2, “Pipe Support Jurisdiction Boundaries,” states that support design jurisdictional boundaries at interfaces with piping, structure, or intervening elements are defined in ASME BPVC, Section III, Subsection NF, paragraph NF-1130, “Boundaries of Jurisdiction.” If piping supports transmit loads to surface-mounted baseplates as discussed in ASME BPVC, Section III, paragraph NF-1132(d), the baseplates are within the building structure jurisdiction. The NRC staff finds that the jurisdictional boundaries between pipe supports and interface attachment points meet the SRP 3.12 guidance and ASME BPVC Section III, Subsection NF requirements.

PSAR Section 3.12.4.3, “Pipe Support Loads and Load Combinations, Table 3.12-5, “Load Combinations and Acceptance Criteria for ASME Class 1, 2, and 3 Snubber Type Supports,” and Table 3.12-6, “Load Combinations and Acceptance Criteria for ASME Class 1, 2, and 3 Rigid Type Supports,” provide load combinations for snubber and rigid pipe supports for ASME Class 1, 2, and 3 piping. The NRC staff finds that the load combinations for ASME Classes 1, 2, and 3 piping systems are consistent with the respective requirements of ASME BPVC, Section III, Subsection NF for the various service levels and therefore are acceptable.

PSAR Section 3.12.4.4, “Pipe Support Baseplate and Anchor Bolt Design,” states that aspects of the anchor bolt design, including baseplate flexibility and factors of safety are used in the development of anchor bolt loads, as addressed in NRC Bulletin 79-02, Rev.2, “Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts,” and per ASME BPVC, Section III, paragraph NF-1132(d). The NRC staff finds that the pipe support baseplate and anchor bolt designs are consistent with the SRP Section 3.12 guidance and therefore are acceptable.

PSAR Section 3.12.4.6, “Use of Snubbers,” states that snubbers are modeled with an equivalent stiffness based on dynamic tests or on data provided from the vendor. The permissible loads that the piping system can place on snubbers are defined in accordance with ASME BPVC, Section III, Subsection NF. The NRC staff finds that defining the snubber dynamic loads per ASME BPVC, Section III, Subsection NF is acceptable and consistent with SRP Section 3.12.

PSAR Section 3.12.4.7, “Pipe Support Stiffness,” states that rigid supports are sized and qualified for a specified deflection limit for all conditions in the restraint directions. Where the support deflection cannot meet the minimum criteria, the actual support stiffness is developed and either justified as sufficiently rigid or included in the piping analysis. In the event the actual stiffness is included, all supports within the stress case boundary are evaluated with actual stiffness values. The NRC staff finds that the pipe support deflection design provisions are consistent with the guidance in SRP Sections 3.12 and 3.9.2.

PSAR Section 3.12.4.9, “Design of Supplementary Steel,” states that the design of structural steel for use as frame type pipe supports are linear supports as defined in ASME Code, Section III, Subsection NF. PSAR Section 3.12.4.10, “Consideration of Friction Forces,” states that the friction force on the pipe during axial movement is considered in the design of the support. PSAR Section 3.12.4.11, “Pipe Support Gaps and clearances,” provides the acceptance criteria for support clearances for frame type supports that allow radial thermal expansion of the pipe without imposing any thermal binding and to retrain the pipe. The NRC staff finds that the design of supplementary steel, consideration of friction forces, and the provisions for pipe support clearances meet the guidance in SRP 3.12 and therefore are acceptable.

PSAR Section 3.12.4.12, “Instrumentation Line Support Criteria,” states that the instrumentation line supports attached to ASME Class 1 piping meet the appropriate requirements of ASME BPVC, Section III, Subsection NF. PSAR Section 3.12.4.13, “Pipe Deflection Limits,” states that Rigid supports are sized and qualified for a specified pipe deflection limit for all conditions in the restraint directions. Where the support deflection cannot meet the minimum criteria, the actual support stiffness is developed and either justified as sufficiently rigid or included in the piping analysis. The NRC staff finds that the instrumentation line support criteria and pipe deflection limits for the support design are consistent with the SRP Section 3.12 guidance and therefore are acceptable.

As discussed above, the design and analysis of the ASME Code Class 1, 2, and 3 piping systems, piping components and their associated supports are consistent with the relevant NRC guidance including SRP 3.12, and therefore meet the relevant requirements of 10 CFR 50.55a, Appendix S to 10 CFR Part 50, and GDC 1, 2, 4, 14, and 15. As such, the design bases for methods of analysis for ASME Code Class 1, 2, and 3 piping systems, piping components and their associated supports for the preliminary design have been explained sufficiently under 10 CFR 50.34(a)(3)(ii).

3.12.4 Conclusion

The staff has reviewed the available information provided in the PSAR and, for the reasons given above, concludes that the description of the design and service combinations of loadings as applied to ASME code Class 1, 2, and 3 piping systems are acceptable and meet the requirements of 10 CFR 50.55a, 10 CFR Part 50, Appendix S, and GDC 1, 2, 4, 14, and 15. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.13 Threaded Fasteners - ASME Code Class 1, 2, and 3

PSAR Section 3.13, “Threaded Fasteners, ASME Code Class 1, 2, and 3,” addresses the application of the requirements in ASME BPVC, Section III, Division 1, as incorporated by reference in 10 CFR 50.55a (TN249) for ASME BPVC Class 1, 2, and 3 pressure-retaining bolts, studs, nuts, and washers (collectively referred to as threaded fasteners) in support of the CPA for CRN-1. The NRC staff evaluation considered the materials selection, mechanical testing, special processes and controls, fracture toughness requirements for ferritic materials, fabrication inspection, quality records, and preservice inspection and ISI requirements for threaded fasteners for CRN-1.

3.13.1 Design Aspects

3.13.1.1 Introduction

PSAR Section 3.13.1, "Design Aspects," indicates that the design of pressure boundary threaded fasteners for CRN-1 will comply with ASME BPVC, Section III, Subsections NB, NCD, NE, NF, and NG, paragraphs NB-3000, NCD-3000, NE-3000, NF-3000, and NG-3000, as incorporated by reference in 10 CFR 50.55a. The applicant stated that load combinations and associated factors will satisfy the guidance in RG 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components." PSAR Section 3.13.1 also indicates that fabrication of threaded fasteners will comply with ASME BPVC, Section III, Subsections NB, NCD, NE, NF, and NG, paragraphs NB-4000, NCD-4000, NE-4000, NF-4000, and NG-4000, and that inspections will comply with paragraphs NB-2500, NCD-2500, NE-5000, NF-2500, and NG-2500, as incorporated by reference in 10 CFR 50.55a.

3.13.1.2 Regulatory Evaluation

The NRC regulations in 10 CFR 50.34, "Contents of Applications; Technical Information," in paragraph (a), Preliminary safety analysis report, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. Paragraph (2) in 10 CFR 50.34(a) requires that the PSAR provide a summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. Paragraph (3) in 10 CFR 50.34(a) requires in item (i) that the PSAR provide a preliminary design of the facility including the principal design criteria for the facility with reference to Appendix A in 10 CFR Part 50 for minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which CPs have previously been issued by the Commission.

The NRC regulations in 10 CFR Part 50, Appendix A, include requirements for threaded fasteners in nuclear power plants. With respect to the threaded fasteners in CRN-1, these requirements include the following GDC:

- GDC 1 – Quality standards and records
- GDC 14 – Reactor coolant pressure boundary
- GDC 30 – Quality of reactor coolant pressure boundary
- GDC 31 – Fracture prevention of reactor coolant pressure boundary

The NRC regulations in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," specify QA requirements for the design, fabrication, construction, testing, and records control of certain SSCs in a nuclear power plant. With respect to threaded fasteners, the NRC regulations in 10 CFR Part 50, Appendix B, require controlling the cleaning of material and equipment to prevent damage or deterioration.

The NRC regulations in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," relate to materials testing and acceptance criteria for fracture toughness of RCPB components. Those regulations specify fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB.

The NRC regulations in 10 CFR 50.55a, "Codes and standards," incorporate by reference the ASME *Boiler and Pressure Vessel Code* (BPVC), Section III, "Rules for Construction of Nuclear

Facility Components," and Section XI, "Rules for Inspection of Nuclear Power Plant Components." These regulatory requirements apply to threaded fasteners for the CRN-1. The selection of materials, design, testing, fabrication, installation and inspection of threaded fasteners and mechanical joints are acceptable if they meet the criteria of the ASME BPVC, Section III, for Class 1, 2, and 3 components. The NRC regulations in 10 CFR 50.55a permit the use of code cases that have been adopted in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," in lieu of certain criteria for ASME BPVC, Section III, Class 1, 2, and 3 components.

RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," provides QA criteria for cleaning fluid systems and associated components that ensure compliance with 10 CFR Part 50, Appendix B. Application of the cleaning criteria in threaded fasteners provides assurance that contaminants to which they could be exposed will not damage or deteriorate the materials, alter their properties, accelerate effects associated with aging, or increase the susceptibility to failure mechanisms such as stress corrosion cracking.

RG 1.65, Revision 1, "Materials and Inspections for Reactor Vessel Closure Studs," provides guidance on reactor vessel closure stud bolting.

In RG 1.70 (Revision 3), the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. Section 5.3.1.7, "Reactor Vessel Fasteners," in RG 1.70 provides guidance for the materials and design of reactor vessel fasteners for nuclear power plant applications. RG 1.70 does not include a Section 3.13 for threaded fasteners.

NRC Standard Review Plan (SRP) Section 3.13, "Threaded Fasteners – ASME Code Class 1, 2, and 3," provides guidance for selection of materials, design, inspection and testing of threaded fasteners.

In its review, the NRC staff evaluated whether the applicant complied with the NRC regulatory requirements in 10 CFR 50.34(a) as applicable for a new CPA with respect to the design of threaded fasteners for CRN-1. The NRC staff technical evaluation of the PSAR Section 3.13.1 with respect to the applicable regulatory requirements and guidance for the design of threaded fasteners is described in the following section of this NRC safety evaluation.

3.13.1.3 Technical Evaluation

The NRC regulations in 10 CFR 50.34, "Contents of applications; technical information," in paragraph (a), Preliminary safety analysis report, require CP applicants to submit a PSAR describing specific aspects of the proposed nuclear power plant for NRC review. In RG 1.70, the NRC provides guidance for preparation of a PSAR describing the nuclear power plant for a CPA. Section 5.3.1.7, "Reactor Vessel Fasteners," in RG 1.70 provides guidance for the materials and design of reactor vessel fasteners for nuclear power plant applications. The NRC staff reviewed PSAR Section 3.13.1 for compliance with the NRC regulations in 10 CFR 50.34(a), and consistency with the NRC guidance in RG 1.70, as applicable to a new CPA, for the content of the PSAR with respect to the design of threaded fasteners for CRN-1. This SE section describes the NRC staff review of the PSAR Section 3.13.1 with respect to the information required by 10 CFR 50.34(a) for the contents of the PSAR as part of a CPA with applicable guidance in RG 1.70 for design of threaded fasteners in CRN-1.

PSAR Section 3.13.1.1 specifies that the RCPB fasteners will be designed, fabricated, and tested to the highest quality standards practical, in accordance with 10 CFR Part 50,

Appendix A, GDC 1 and 30, to have an extremely low probability of abnormal leakage or rapidly propagating failure or gross rupture, in accordance with GDC 14. In accordance with GDC 31, the design of CRN-1 threaded fasteners will include sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a nonbrittle manner, and the probability of rapidly propagating fracture is minimized. This evaluation will reflect consideration of service temperatures, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining material properties; the effects of irradiation on material properties; residual, steady state and transient stresses; and size of flaws.

The design of the threaded fasteners will comply with the NRC regulations in 10 CFR Part 50, Appendix A, as follows:

- GDC 14 – “Reactor Coolant Pressure Boundary,” as it relates to the RCPB being designed, fabricated, erected, and tested in a manner that provides assurance of an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture by compliance with 10 CFR Part 50, Appendix G.
- GDC 30 – “Quality of Reactor Coolant Pressure Boundary,” as it relates to components of the RCPB being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety category function to be performed by compliance with ASME BPVC, Section III, as incorporated by reference in 10 CFR 50.55a.
- GDC 31 – “Fracture Prevention of Reactor Coolant Pressure Boundary,” as it relates to the requirement that the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner, and the probability of rapidly propagating fracture is minimized with compliance with ASME BPVC, Section III, as incorporated by reference in 10 CFR 50.55a, and 10 CFR Part 50, Appendix G.

The threaded fasteners at CRN-1 will satisfy the QA criteria in 10 CFR Part 50, Appendix B.

PSAR Section 3.13.1 references ASME BPVC, Section III, Subsection NCD, paragraph NCD-2300, with respect to ferritic fasteners in ASME Class 2 and 3 systems. PSAR Section 13.3.1 indicates that 10 CFR Part 50, Appendix G, will be applied to threaded fasteners as it relates to materials testing and acceptance criteria for fracture toughness of RCPB components.

The NRC regulations in 10 CFR 50.55a incorporate by reference the design criteria of ASME BPVC, Section III, for Class 1, 2, and 3 components. The selection of materials, design, testing, fabrication, installation and inspection of threaded fasteners and mechanical joints will be acceptable if they meet the criteria of the ASME BPVC, Section III, for Class 1, 2, and 3 components. Case cases may be applied as accepted by the NRC in RG 1.84, in lieu of applicable criteria in ASME BPVC, Section III, for Class 1, 2, and 3 components.

PSAR Section 3.13.1 specifies that the design of threaded fasteners (e.g., bolts and studs) will comply with ASME BPVC, Section III, Subsections NB, NCD, NE, NF, and NG, paragraphs NB-3000, NCD-3000, NE-3000, NF-3000, and NG-3000. The load combinations and associated load factors will satisfy the guidance in RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.” Fabrication of threaded fasteners will comply with ASME BPVC, Section III, Subsections NB, NCD, NE, NF, and NG, paragraphs

NB-4000, NCD-4000, NE-4000, NF-4000, and NG-4000. Inspection of threaded fasteners will comply with ASME BPVC, Section III, Subsections NB, NCD, NE, NF, and NG, paragraphs NB-2500, NCD-2500, NE-5000, NF-2500 and NG-2500.

Material used for threaded fasteners (e.g., bolts and studs) will comply with the requirements of ASME BPVC, Section III, Subsection NCA, and Subsections NB, NCD, NF, and NG, paragraphs NB-2128 (Class 1), NCD-2128 (Class 2 and 3), NF-2128 (supports), or NG-2121 (core support structures). Material for nuts will conform to SA-194 or to the requirements of one of the specifications for nuts. Material for bolting will conform to ASME BPVC, Section II, Part D, Subpart 1, Table 4 (Class 1) or Table 3 (Class 2 and 3). Fracture toughness testing will be performed in accordance with ASME BPVC, Section III, Subsections NB, NCD, NF, and NG, paragraphs NB-2300, NCD-2300, NF-2300, or NG-2300. RG 1.65, Revision 1, “Materials and Inspections for Reactor Vessel Closure Studs,” provides guidance on reactor vessel closure stud bolting.

PSAR Section 3.13.1 references PSAR Sections 4.2 and 4.6 for the use of threaded fasteners for tie rods lower end plugs, upper tie plate to accept channel fastener bolt, and for the FMCRD middle flange attachments to CRD housing.

PSAR Section 3.13.1 references PSAR Section 5.2.3 for specifications pertaining to pressure-retaining ferritic materials, nonferrous metals, and austenitic stainless steels, including bolting and weld materials, which are used for each component (vessels, piping, and valves) of the RCPB. The adequacy and suitability of the ferritic materials, stainless steels, and nonferrous metals specified for these applications are specified in PSAR Sections 5.2.3.1 through 5.2.3.4. PSAR Tables 5.2-2 and 5.2-3 list the materials and ASME standards for the RCPB which include the pipe, body bolting and bolting nuts.

PSAR Section 3.13.1 specifies that the use and selection of lubricants or sealants for threaded fasteners will conform to the practices provided in NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants.”

Criterion XIII in 10 CFR Part 50, Appendix B, requires that measures be established to control the cleaning of material and equipment to prevent damage or deterioration. RG 1.28, “Quality Assurance Program Criteria (Design and Construction),” provides QA criteria for cleaning fluid systems and associated components that ensure compliance with 10 CFR Part 50, Appendix B. Application of the cleaning criteria in threaded fasteners provides assurance that contaminants to which they could be exposed will not damage or deteriorate the materials, alter their properties, accelerate effects associated with aging, or increase the susceptibility to failure mechanisms such as stress corrosion cracking. Application of these criteria reduces the likelihood that degradation of threaded fasteners could lead to loss of pressure boundary integrity.

For verification of conformance to the applicable ASME BPVC requirements, a chemical analysis will be performed for each heat of material. Testing for mechanical properties will be performed on samples representing each heat of material and, where applicable, each heat treat lot.

ASME BPVC Class 2 and 3 pressure-retaining ferritic material and its welded material will be impact tested in accordance with the requirements of ASME BPVC, Section III, Division 1, Subsection NCD, paragraphs NCD-2300 and NCD-2400, to ensure adequate fracture toughness properties. PSAR Sections 5.3.1.5 and 5.3.2 discuss the fracture toughness and the pressure-temperature limits of the RCPB and reactor vessel.

The fracture toughness of ferritic bolts, studs, and nuts (i.e., made from either low-alloy steel or carbon steel materials) is acceptable if the ASME BPVC, Section III, criteria are specified for ASME BPVC Class 1, 2, and 3 systems. Ferritic bolts, studs, and nuts used in RCPB applications also will meet the fracture toughness requirements of 10 CFR Part 50, Appendix G, for the supplemental requirements for ASME BPVC Class 1 systems.

Fabrication of threaded fasteners will comply with ASME BPVC, Section III, Division 1, Subsection NB, paragraph NB-4000 for Class 1 components; Subsection NCD, paragraph NCD-4000 for Class 2 and 3 components; Subsection NE, paragraph NE-4000; Subsection NF, paragraph NF-4000; and Subsection NG, paragraph NG-4000.

Inspection of threaded fasteners will comply with ASME BPVC, Section III, Division 1, Subsection NB, paragraph NB-2500 for Class 1 components; Subsection NCD, paragraph NCD-2500 for Class 2 and 3 components; Subsection NE, paragraph NE-2500; Subsection NF, paragraph NF-2500; and Subsection NG, paragraph NG-2500.

The visual examinations will be performed in accordance with procedures qualified to ASME BPVC, Section III, Division 1, Subsection NB, paragraph NB-5100 for Class 1 components; Subsection NCD, paragraph NCD-5100 for Class 2 and 3 components; Subsection NE, paragraph NE-5100; Subsection NF, paragraph NF-5100; and Subsection NG, paragraph NG-5100.

The visual examinations will be performed by personnel qualified in accordance with ASME BPVC, Section III, Division 1, Subsection NB, paragraph NB-5500 for Class 1 components; Subsection NCD, paragraph NCD-5500 for Class 2 and 3 components; Subsection NE, paragraph NE-5500; Subsection NF, paragraph NF-5500; and Subsection NG, paragraph NG-5500.

Bolts, studs, and nuts will be visually examined for discontinuities including cracks, bursts, seams, folds, thread lap, voids, and tool marks.

For threaded fasteners, documentation related to fracture toughness (as applicable) and Certified Material Test Reports (CMTRs) will be provided as part of the ASME BPVC, Section III, records at the time the parts are shipped. This documentation will be part of the required records that are maintained at the site in accordance with ASME BPVC, Section III, Subsection NCA, paragraph NCA-1224, and 10 CFR 50.70, "Inspections."

Based on its review, the NRC staff finds that PSAR Section 3.13.1 provides acceptable information related to the design of threaded fasteners that meets 10 CFR 50.34(a)(2) and 10 CFR 50.34(a)(3)(i) consistent with applicable guidance in RG 1.70 to support issuance of the CP for CRN-1.

3.13.1.4 Conclusion

Based on its review, the NRC staff concludes that the information in PSAR Section 13.3.1 for the design of threaded fasteners is acceptable to support the CPA for CRN-1. Therefore, the NRC staff concludes that the information in PSAR Section 13.3.1 is sufficient and meets the regulatory requirements of 10 CFR 50.34(a) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the NRC regulations of 10 CFR 50.35.

3.13.2 Preservice and In-Service Inspection Requirements

PSAR Section 3.13.2, "Preservice and Inservice Inspection Requirements," references PSAR Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing," and PSAR Section 6.6, "Preservice and Inservice Inspection of Class 2 and Class 3 Components," for the description of the Preservice Inspection and ISI programs applicable to threaded fasteners in CRN-1. The NRC review of those PSAR sections is described in the applicable sections of this report.

3.13.3 Conclusion

Based on its review of the information provided in PSAR Section 13.3, and for the reasons given above, the NRC staff concludes that the preliminary design for the ASME BPVC Class 1, 2, and 3 threaded fasteners is acceptable and meets the relevant requirements of 10 CFR Part 50, Appendix A, GDC 1, 4, 14, 30, and 31; 10 CFR Part 50, Appendix B, Criterion XIII; and 10 CFR Part 50, Appendix G. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements 10 CFR 50.34(a) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.

3.14 3A-N Appendices

3.14.1 3A PRELIMINARY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.14.1.1 3A.1 Introduction

See Sections 3.2.3 and 3.2.4 of this report.

3.14.1.2 3A.2 Regulatory Evaluation

N/A

3.14.1.3 3A.3 Technical Evaluation

See Sections 3.2.3 and 3.2.4 of this report.

3.14.1.4 3A.4 Conclusion

See Sections 3.2.3 and 3.2.4 of this report.

3.14.2 3B INTEGRATED REACTOR BUILDING FINITE ELEMENT MODEL

3.14.2.1 3B.1 Introduction

PSAR Appendix 3B, submitted by CPA Supplement 1 dated October 1, 2025 (TVA 2025-TN13050), describes, illustrates and validates the detailed three-dimensional (3-D) finite element (FE) models developed for the one-step structural analyses (including seismic soil-structure interaction [SSI]) for the design of the integrated reactor building (IRB) structures consisting of the Reactor Building structure outside containment, the containment, containment internal structures and their common mat foundation. PSAR Appendix 3B states that these models are developed implementing the analysis procedures presented in PSAR Sections 3.7.2.3 and 3.8.4.1.4.

The applicant amended PSAR Appendix 3B by CPA Supplement 4 dated December 2, 2025 (TVA 2025-TN13044) and CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029).

3.14.2.2 3B.2 Regulatory Evaluation

10 CFR 50.34(a)(4) (TN249) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures of the facility including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures and components provided for the prevention of accidents and the mitigation of the consequences of accidents. The 3-D FE models described in this PSAR Appendix 3B are used to implement this preliminary analysis and design evaluation of the IRB structures, in part, to determine structural demands under design loads and load combinations by implementing the design analysis procedures in PSAR Sections 3.7.2.3 and 3.8.4.1.4 which the staff evaluated and found acceptable in Sections 3.7.2.3.3 and 3.8.4.3.1.4 of this report. The staff review is therefore against the design analysis procedures in PSAR Sections 3.7.2.3 and 3.8.4.1.4.

SRP Sections 3.7.2.II.3, 3.7.2.II.4 and 3.8.4.II.4, against which PSAR Sections 3.7.2.3 and 3.8.4.1.4 were evaluated, provide guidance for dynamic modeling of structures for seismic analysis and structural modeling for detailed design analysis of all applicable load combinations, including assumptions on boundary conditions. Since the applicant is using a one-step analysis approach for CRN-1 IRB structures, the same 3-D FE models are used to perform the dynamic SSI analyses for seismic response and structural design analyses of the IRB structures, with exception of models used for analysis of subsystems.

3.14.2.3 3B.3 Technical Evaluation

The staff reviewed PSAR Sections 3B.1 through 3B.6, as amended by CPA Supplements 4 and 5, against the design analysis procedures in PSAR Section 3.7.2.3 for seismic analyses and PSAR Section 3.8.4.1.4 for structural analyses of the integrated RB structures. The staff evaluated and found acceptable these analyses methodology in Sections 3.7.2.3.3 and 3.8.4.3.1.4 of this report, respectively.

From its review of PSAR Sections 3B.1 through 3B.6, as amended, the staff finds that the applicant has developed adequate 3-D FE models for seismic SSI analyses and structural analyses for preliminary analysis and design evaluation of the IRB structures for the following reasons:

- The 3-D FE models described and illustrated in PSAR Sections 3B.1 through 3B.5 thus demonstrate an adequate implementation of the analysis methodology and procedures presented in Section 3.7.2.3 and Section 3.8.4.1.4 which the staff evaluated and found acceptable in Sections 3.7.2.3.3 and 3.8.4.3.1.4 of this report, respectively.
- PSAR Section 3B.1 and Figures 3B-1 and 3B-2 describe and illustrate the IRB 3-D FE model, including the SCCV containment penetrations, used to perform the various structural analyses discussed in PSAR Section 3.8.4.1.4. The origin for this model is at the center of the IRB at grade level with X, Y representing the horizontal axes and Z the vertical axis. The

RPV and internals and CEPSS are integrated into the IRB FE model using a lumped mass stick (LMS) model developed as described in PSAR Section 3.7.3.3.3 and is illustrated in PSAR Figure 3B-3. As illustrated in the above referenced figures, this integrated model is developed using appropriate element types to represent the mass, stiffness, damping characteristics of the integrated Reactor Building structures for the analyzed loads being evaluated and resulting stress responses. The model has a sufficiently refined mesh to accurately capture the dynamic response of the structures over the frequency range of interest and to enable accurate calculation of structural stress demands. Mesh refinements in areas of geometric structural discontinuities are used, as applicable.

- Appropriate material properties indicated in PSAR Tables 3B-2 and 3B-3 are used for steel-plate composite (DP-SC or SC [steel-plate composite]) and steel elements in the model. The DP-SC components (configurations shown in PSAR Figure 3B-4) are modeled using elastic shell elements with equivalent thickness, elastic modulus, and material density calibrated to match the DP-SC effective stiffness and mass properties are calculated using procedures in Section 5.5 of NRC-approved LTR NEDC-33926P-A. Effective material properties used for the operating conditions represent the best estimate (BE) structure stiffness and effective material properties for the fully cracked (accident) conditions are referred to as the Lower Bound (LB) structure stiffness are appropriately assigned. Additional sacrificial faceplate thickness or liners used for corrosion protection are appropriately not considered in the calculation of FE equivalent section properties.
- The RB FE model utilizes centerline shell element models of DP-SC or SC components where adjacent or intersecting component elements share nodes at their intersecting lines to simulate essentially rigid behavior of welded connections. The RPV and CEPSS connections are appropriately modeled with interfacing components.
- The IRB model is analyzed for the design loads described in PSAR Section 3.8 (and indicated in Table 3B-1 as amended by CPA Supplement 4), which are assigned to the model as static and dynamic masses as described in PSAR Section 3B.1.7.
- Section 3B.2 describes the standalone Case 1 3-D FE model developed for the one-step seismic soil-structure interaction (SSI) analyses of the deeply embedded integrated Reactor Building structures and the Case 2 3-D FE dynamic model developed for SSSI analyses of the deeply embedded integrated RB structures and the surface-founded power block structures. The Case 1 and Case 2 dynamic seismic SSI models are conservatively assigned BE structure stiffnesses with seismic and dead load inertial masses and lower OBE-level damping properties.
- The Case 1 3-D FE model, illustrated in PSAR Figures 3B-5 and 3B-6, consists of the FE representations of the IRB shown in PSAR Figure 3B-1 and the surrounding subgrade described in PSAR Section 3.7.2, including the explicit representation of certain construction elements in the CRN-1 excavation plan (i.e., the mud mat under the Reactor Building mat foundation, the concrete fill in the annulus and shelf around the Reactor Building exterior wall. The model also includes solid elements that may be assigned material properties corresponding to soil or simulating the temporary excavation support to assess their effects on the RB seismic response discussed in PSAR Sections 3.7.2.3 and 3.7.2.9.2. This model is paired with the range of in situ, as-built subgrade conditions discussed in PSAR Sections 3.7.1.3 to perform the Case 1 seismic SSI analysis, results of which is summarized in PSAR Appendix 3C.
- In the combined Case 2 3-D FE model shown in PSAR Figure 3B-7, coarse surface mounted FE models representing the global dynamic properties and weight of the surrounding power block structures and foundations (foundations centerline at

grade level; turbine generator pedestal isolated model as shown in PSAR Figure 3B-9) are coupled with the IRB and subgrade model shown in PSAR Figure 3B-5 to capture the structure-soil-structure interaction (SSSI) effects described in PSAR Section 3.7.2.4. This combined SSI model (Case 2 model) is paired with a second case of as-built subgrade conditions where the in-situ soil and weathered rock are excavated down to competent rock and replaced with engineered fill up to plant grade elevation 814.5 ft to provide adequate support for the adjacent power block structures. The results of the Case 2 seismic SSI analysis are also summarized in PSAR Appendix 3C.

- The soil and soil-structure interfaces are adequately modeled in the Case 1 and Case 2 dynamic SSI analyses models in SASSI as follows:
  - Far-field subgrade model representing the properties of the subgrade materials that are approximated as continuum infinite horizontal layers with uniform linear elastic properties resting on surface of uniform elastic half-space. A viscous lower boundary of the subgrade model is established at a depth of 482 ft consistent with the guidance in SRP Section 3.7.2.II.4.
  - Near-field subgrade model, illustrated in PSAR Figure 3B-9, representing the properties of subgrade materials supporting the adjacent Power Block foundations and around the RB below grade exterior wall. It consists of a single layer of near-field solid elements assigned subgrade material properties placed below the power block foundations, and the solid FE elements around the below grade portion of the RB exterior wall that represent the properties of the concrete backfill. The FE of the excavated soil volume and near-field volume shares the same nodes at the excavated volume perimeter.
  - Excavated soil volume model, illustrated in PSAR Figure 3B-10, consisting of 3-D solid FEs representing the properties of the volume of far-field subgrade replaced by the RB and adjacent near-field volume elements.
  - Linear spring elements (three at each node) to connect the near-field subgrade elements to structural elements, which are assigned stiffness properties that can be adjusted to simulate different interface conditions. The excavated volume and the near-field volumes are also connected with linear spring elements at the near-field (backfill) perimeter.
  - The soil elements used in dynamic and subgrade impedance calculations use a consistent layer thickness governed by the dynamic response which is sufficiently refined to transmit the entire frequency range of interest.
- PSAR Sections 3B.3, 3B.4 and 3B.5 describe the appropriate 3-D FE models used, respectively, for the 1-g static SSI analyses. for static and quasi-static analyses, and for thermal stress analysis, which the staff found reasonable and rational.
- PSAR Table 3B-1, as amended by CPA Supplement 4, summarizes all the various IRB analyses cases and corresponding model requirements by specifying for each analysis case (or loading type) the applicable design load(s) applied in the model, the analysis method (including computer program used), and the appropriate structural stiffness, subgrade properties (with UB SSI analysis profiles appropriately used in conjunction with compatible LB damping values, and vice versa), the contact spring stiffness of the embedded Reactor Building exterior wall and mat foundation interfaces with the subgrade (boundary conditions), and the specific structural 3-D FE model developed in Appendix 3B used which the staff found to be rational and appropriate.
- The FE models are developed using ANSYS computer program and for seismic SSI and SSSI analyses translated into SASSI computer program. The structural analyses and

seismic analyses are performed using verified and validated ANSYS and SASSI computer programs, respectively. The description and verification and validation of these programs are provided in PSAR Appendix 3I, which the staff evaluated and found acceptable in Section 3I of this report.

- PSAR Section 3B.6 documents the verification of the IRB seismic analyses models and their translation from ANSYS to SASSI by performing the following steps: Modal analysis of the Reactor Building fixed base model to understand the key dynamic properties of the model. The resulting IRB significant mode frequencies and mass participation ratios shown in PSAR Table 3B-4 are reasonable and indicated to agree reasonably with simplified hand calculations.
- 1-g analysis of the Reactor Building fixed base model in X-, Y-, and Z-direction to assess the total seismic mass and to examine deformed shape and ensure component connectivity in the three directions. The seismic mass and maximum displacements shown in PSAR Table, and the deformed shapes in the three directions shown in PSAR Figures 3B-12 through 3B-14 are reasonable and as expected confirm element connectivity between various structural components.
- Modal analysis of the isolated RPV LMS model for verification of adaptation into the IRB model. This was done by comparing the dynamic characteristics of the RPV lumped mass stick (LMS) model "isolated" within the IRB model (by applying fixed boundary conditions at appropriate connection points with the pedestal) with benchmark results of the eigen value analysis of the standalone RPV LMS model. The resulting modal properties (frequencies and cumulative mass participations) in PSAR Figure 3B-15 show agreement and demonstrate the coupled dynamic response between the integrated RPV LMS model and the Reactor Building is adequately captured.
- Modal analysis of the Power Block structures for verification of adaptation into the IRB model. This is done by comparison of the dynamic frequencies of the primary vibration modes in each direction between the provided coarse mesh models in the combined seismic SSI analysis IRB model and each power block building "isolated" in the global structural model with those from refined standalone models of the power block structures. The applicant reported that results of the dynamic characteristics (total mass, dominant frequencies) of the coarse dynamic models match reasonably well with those from refined models; therefore, indicating the coarse FE models of the Power Block structures provide an adequate representation of the global dynamic properties for the SSI analyses of the IRB.
- Transfer function comparison between ANSYS and SASSI RB fixed base models for verification of translation of the IRB model to SASSI by comparing the total acceleration transfer functions from the ANSYS fixed-base Reactor Building model (determined from harmonic analysis) and the SASSI fixed base model. The transfer functions are defined as the ratio of the acceleration at a particular point in the structure to the amplitude of the input acceleration at the base. Comparison of pairs of transfer functions (horizontal and vertical directions) at two locations shown in PSAR Figures 3B-16 and 3B-17 indicate good agreement between transfer functions from ANSYS and SASSI. The figures also show that the peaks occur at frequencies consistent with the primary mode frequencies from modal analysis, confirming the translation of each model from ANSYS to SASSI.

3.14.2.4 3B.4 Conclusion

Based on the above review of PSAR Appendix 3B as amended, the staff concludes that the applicant has developed reasonable 3-D FE models and analyses cases for seismic SSI and SSSI analyses and structural analyses of the IRB structures by adequately implementing the design analysis procedures described in PSAR Sections 3.7.2.3 and 3.8.4.1.4, which the staff evaluated and found acceptable these analyses methodology in Sections 3.7.2.3.3 and 3.8.4.3.1.4 of this report, respectively. The staff also concludes that these models are adequate to perform the preliminary analysis and design evaluation of the IRB structures pursuant to the regulatory requirements of 10 CFR 50.34(a)(4) and adequately supports the issuance of a CP pursuant to the regulations in 50.35. The results of these preliminary analyses and design evaluation are reported for selected IRB structures in PSAR Appendices 3C, 3E through 3H, the staff evaluation of which are provided in the corresponding sections of this report.

3.14.3 3C SEISMIC SOIL STRUCTURE INTERACTION ANALYSIS RESULTS

3.14.3.1 3C.1 Introduction

PSAR Appendix 3C, submitted by CPA Supplement 1 dated October 1, 2025 (TVA 2025-TN13050), documents the implementation and results of the preliminary one-step seismic SSI analyses and sensitivity analyses of the integrated Reactor Building (IRB) using the standalone IRB model (Case 1 subgrade profile) and combined IRB with Power Block structures model (Case 2 subgrade profile) described in PSAR Appendix 3B, Section 3B.2.

The applicant provided an amended version of Appendix 3C in CPA Supplement 7 dated March 2, 2026 (TVA 2026-TN13043) and in Revision 1 of the PSAR.

3.14.3.2 3C.2 Regulatory Evaluation

10 CFR 50.34(a)(4) (TN249) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures of the facility including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

The results of the seismic SSI and sensitivity analyses in PSAR Appendix 3C are used to implement this preliminary analysis and design evaluation of the IRB structures, in part, to determine seismic structural demands under design basis seismic SSE loads and load combinations by implementing the seismic design basis and analysis procedures in PSAR Sections 3.7.1 and 3.7.2, which the staff evaluated and found acceptable in Sections 3.7.1.3 and 3.7.2.3 of this report.

3.14.3.3 3C.3 Technical Evaluation

The staff reviewed PSAR Appendix 3C, as amended, to confirm/verify that the implementation of the preliminary seismic SSI analyses to determine seismic structural demands are consistent with the seismic design basis and seismic analysis methodology/procedures in PSAR Sections 3.7.1.3, 3.7.2.3, 3.7.2.4, and 3.7.2.9, which the staff evaluated and found acceptable in corresponding subsections in Sections 3.7.1.3 and 3.7.2.3 of this report, and that the analyses results are reasonable.

The staff reviewed the implementation and results of the SSI analyses in PSAR Sections 3C.1 through 3C.9, and noted or confirmed the following:

- The Case 1 and Case 2 seismic analyses are performed using the 3-D FE standalone and combined models, described in PSAR Section 3B.2, following the one-step linear analysis approach discussed in PSAR Section 3.7.2.1. The approach allows seismic stress demands to be obtained directly from the results of the linear elastic SSI analyses performed using SASSI computer program. In both cases, the design basis SSE control motion (shown as FIRS in PSAR Figure 3C-4) is applied to the model in the free field at the level of the bottom of the Reactor Building foundation.
- Shell element forces and moments from the Case 1 and Case 2 SSI analyses reported at the center of the element are the element stress demands. The co-directional stress demands are calculated by algebraically combining for each element the time histories of responses due to the three earthquake components using the time-step-by-time-step approach discuses in PSAR Section 3.7.2.6. The combined stress demands are then transformed from the local element co-ordinate system to the global coordinate system for use as input for the design of structural members.
- *Design Stress Demand Calculation:* Average stress demand results from the corresponding five sets of acceleration time histories (ATHs) represent the response for each of the SSI analysis cases. Enveloped stress demand results obtained from the analyses of the three subgrade profiles (BE, LB, UB) for Case 1 and Case 2 and from the sensitivity analyses provide design stress demands that account for variations and uncertainties.
- *Design In-Structure Response Spectra (ISRS) Calculation:* From PSAR Figures 3C-1 and 3C-2, ISRS are calculated at equally distributed points in the frequency range from 0.1 Hz to 100 Hz. For each of the five sets of ground motion ATHs, the ISRS from the analysis of each of the three earthquake components are combined to get total co-directional response using the SRSS method. For each subgrade profile case (Case 1 and Case 2), the response spectra from the corresponding sets of five ATHs are averaged as illustrated in PSAR Figure 3C-1. The averaged ISRS obtained from the Case 1 and Case 2 SSI analysis cases are then enveloped and the peaks of the enveloping ISRS are broadened by 15 percent to get the design ISRS for each location, as illustrated in PSAR Figure 3C-2. Results of the sensitivity analyses in PSAR Section 3C.7 are also incorporated into the enveloped ISRS if they exceed the Case 1 and Case 2 SSI analysis by more than 10 percent (example at center of mat foundation illustrated in PSAR Figure 3C-25). The staff notes that the above process implemented for the development of the ISRS is consistent with the methodology in PSAR Section 3.7.2.5.1. The staff also observes that the shape, amplitude and frequency content broadened design ISRS developed for the center of the basemat foundation for 4 percent damping, as shown in PSAR Figure 3C-2, are reasonable.
- *Key nodal locations* (total 8), selected per criteria in Section 5.3.1 in LTR NEDO-33914-A, listed in PSAR Table 3C-1 used for reporting responses for verification and validation of the IRB SSI analyses responses are reasonable because they cover a range of critical locations of the IRB from basemat to the roof. Furthermore, examination of enveloped seismic relative displacements at the key nodal locations reported in PSAR Table 3C-3, from the preliminary seismic analysis cases, indicate they are reasonable and small consistent with expected structural behavior. The maximum horizontal displacements of 0.95 inch and 0.76 inch in the horizontal X and Y directions, and 0.55 inch in the vertical direction are at the center of the Reactor Building roof as expected.
- *Key Structural Members* (total 6, namely basemat slab, SCCV top slab, SCCV wall, RPV pedestal, Reactor Building exterior wall, and wing wall), listed in PSAR Table 3C-2 and

illustrated in PSAR Figure 3C-3, selected for reporting, comparison and evaluation of seismic stress/force demands or seismic response for different analyses cases are reasonable because they are selected based on critical support function of safety equipment, critical importance within seismic lateral force resisting system, or critical location along seismic load path.

- *Seismic Analyses Cases* performed (total of 12) presented in PSAR Table 3C-3 and adequately described in PSAR Section 3C.4 are reasonably comprehensive and address effects of uncertainties and variations in parameters on seismic responses. They consist of BE, LB, and UB for each of Case 1 and Case 2 subgrade profile models (six primary design basis cases) and six sensitivity analyses cases using combined SSI FE model and Case 2 BE profile. The sensitivity cases consist of cracked (stiffness/damping variation), excavation support, no-friction interfaces, dry (ground water variations), soil separation, and lean concrete (backfill strength). The sensitivity cases are consistent with the methodology described for effects of parameter variations in PSAR Section 3.7.2.9, which the staff evaluated and found acceptable in Section 3.7.2.3.9 of this report.
- *Selected frequencies of analysis*: The seismic SSI analysis cases are performed for frequencies that adequately characterize the response of the full SSI system. PSAR Section 3C.5 determined, based on the selection criteria in Section 5.3.2 of NRC-approved LTR NEDO-33914-A, the following cutoff frequencies adopted for the analyses: 35 Hz for LB Cases 1 and 2, 51 Hz for the BE Cases 1 and 2, and 70 Hz for UB Cases 1 and 2. The six primary seismic SSI analysis cases are performed using SASSI frequency domain analysis for frequencies starting at 0.012 Hz to the respective cutoff frequency above.
- Results of the sensitivity analysis are included (enveloped) in the Reactor Building seismic design basis when response comparisons show significant exceedances (greater than 10 percent) in the response due to any of the examined effects. Results of the sensitivity evaluations discussed and illustrated in PSAR Sections 3C.7.1 through 3C.7.5 demonstrate that the seismic forces and moments do not differ significantly from the base case (BE Case 2) with some exceptions, notably for the No-Friction (increase) and Cracked (decrease) sensitivity cases.
- Seismic structural stress (force, moments) response characteristics of main stress components in key structural members for Cases 1 and 2 (LB, BE, UB) described in PSAR Section 3C.6.1 and illustrated in PSAR Figures 3C-10 through 3C-21 indicate reasonable and expected structural behavior for the different subgrade cases. Due to the cylindrical shape of the RB exterior wall, the out-of-plane dynamic pressure forces applied on the below grade wall are primarily resisted by hoop membrane forces and in-plane shear forces, and out-of-plane moments are small, Also, the LB case is found to provide the largest seismic demands which is attributed to larger dynamic pressures applied to the deeply embedded structure supported by softer soil layers.
- Critical SSE structural stress demands (membrane forces, moments) obtained for key structural members of containment (SCCV top slab, SCCV wall), RPV pedestal and Reactor Building exterior wall illustrated in PSAR Figures 3C-56 through 3C-61 appear reasonable.
- In PSAR Section 3C.6.3, as amended by CPA Supplement 7 dated March 2, 2026 (TVA 2026-TN13043), the applicant reported the minimum ground contact ratio value from the implementation of linear SSI analyses cases to show conformance with the SRP 3.7.2.II.4 criteria, that linear SSI analysis methods are acceptable if the ground contact ratio, calculated from the linear SSI analysis using the minimum basemat area that remains in compression with the soil, is equal to or greater than 80 percent or 0.8. The staff noted that the reported values of the minimum ground contact ratio from implementation of linear SSI

analyses in PSAR Appendix 3C, as amended in Rev 1 of the PSAR, was 86.6 percent or 0.866, which is greater than 0.8. The staff thus verified/confirmed that implementation of linear SSI analyses remains valid for CRN-1 seismic category I IRB structures.

Based on the above review, the staff finds that that the applicant has demonstrated in PSAR Appendix 3C, as amended, an adequate implementation of the preliminary seismic SSI analyses to determine seismic structural demands consistent with the seismic design basis and seismic analysis methodology in PSAR Sections 3.7.1.3, 3.7.2.3, 3.7.2.4, and 3.7.2.9, which the staff evaluated and found acceptable in corresponding subsections in Sections 3.7.1.3 and 3.7.2.3 of this report. The staff also finds that the results of the SSI analyses performed are reasonable, with a confirmatory operating license action item A-OL-3.7-1.

3.14.3.4 3C.4 Conclusion

Based on the above review of PSAR Appendix 3C, as amended, the staff concludes that the applicant has demonstrated adequate implementation of the preliminary seismic SSI analyses of the IRB structures to determine design seismic structural demands because it is consistent with the seismic design basis and seismic analysis methodology in PSAR Sections 3.7.1.3, 3.7.2.3, 3.7.2.4, and 3.7.2.9. The staff also concludes that the preliminary SSI analyses results are reasonable and adequate to perform the preliminary design evaluation of the IRB structures pursuant to the requirements of10 CFR 50.34(a)(4) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35. The results of these preliminary analyses and design evaluation are reported for selected IRB structures in PSAR Appendices 3F through 3H, the staff evaluation of which are provided in the corresponding sections of this report.

3.14.4 THREE-DIMENSIONAL INTERACTION EVALUATIONS RESULTS

PSAR Appendix 3-D is intended to provide interaction evaluations results of the Seismic Category II and RW-IIa Power Block structures and foundations with the Seismic Category I structures and foundations under extreme loading. PSAR Appendix 3D, submitted in CPA Supplement 1 dated October 1, 2025, states that Implementation of the methodology for these evaluations discussed in PSAR Section 3.3 and Section 3.7 requires the design of the surrounding Seismic Category II and RW-IIa Power Block structures and foundations to be completed to provide quantifiable results, and, thus, the interaction evaluations will be provided to support the FSAR.

The staff will therefore review the interaction evaluation results of the surrounding seismic category II and RW-IIa power block structures with seismic category I RB (under extreme loading) during the FSAR (operating license application) review. The staff finds it reasonable for the applicant to provide this information in the FSAR since they involve power block structures which are non-seismic category I and less safety-significant. The staff will track this as OLA action item A-OL-3D-1.

3.14.5 3E RESPONSE TO STATIC LOADS

3.14.5.1 3E.1 Introduction

PSAR Appendix 3E, submitted by CPA Supplement 1 dated October 1, 2025 (TVA 2025-TN13050), documents the implementation of 1-g static SSI, static and quasi-static and thermal analyses performed for the applicable loads (other than dynamic seismic) to determine structural demands for the design of the integrated Reactor Building (RB) structures. The PSAR states the analyses are performed following the methodology presented in PSAR Section 3.8.4.1.4, using the Finite Element (FE) models described PSAR Section 3B.3 through Section 3B.5, and design parameters discussed in PSAR Section 3E.1.

The PSAR also states that the preliminary analyses presented in the appendix excluded construction, refueling and piping reaction, and tornado missile loadings that were judged to be non-governing but will be considered in the final design of the structures. Beyond design basis events are also not included in the load cases analyzed.

The applicant amended PSAR Appendix 3E by CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029).

3.14.5.2 3E.2 Regulatory Evaluation

10 CFR 50.34(a)(4) (TN249) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures of the facility including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

The results of the 1-g static SSI, static and quasi-static and thermal analyses presented in PSAR Appendix 3E are used to implement the preliminary analysis and design evaluation of the IRB structures, in part, to determine structural demands under design basis loads and load combinations, other than dynamic seismic, by implementing the applicable design analysis procedures in PSAR Section 3.8.4.1.4, which the staff evaluated and found acceptable in Section 3.8.4.3.1.4 of this report.

3.14.5.3 3E.3 Technical Evaluation

The staff reviewed PSAR Appendix 3E, as amended, to confirm/verify that the implementation of the preliminary 1-g static SSI, static and quasi-static and thermal analyses to determine structural demands for the IRB structures are consistent with the design analysis methodology/procedures in PSAR Section 3.8.4.1.4, which the staff evaluated and found acceptable in Section 3.8.4.3.1.4 of this report, and that the analyses results are reasonable.

The staff reviewed the implementation and results of the 1-g static SSI, static and quasi-static and thermal analyses in PSAR Sections 3E.1 through 3E.4, and noted or confirmed the following:

- The following design parameters were considered and appropriate site-specific input values were used: equivalent linear static subgrade properties (PSAR Table 3E-1 consistent with PSAR Tables 3.8-8 and 3.8-9); additional horizontal rock pressure; ambient temperature; snow loads; wind and extreme wind parameters; pool water levels; crane loading (PSAR Table 3E-2); containment internal normal and accident pressure (PSAR Table 3E-3); internal pressure loading outside containment; internal flooding; summer/winter thermal conditions inside and outside IRB (PSAR Table 3E-4).
- 1-g static SSI analyses are performed using ANSYS computer program and model described in PSAR Section 3B.3. Two sets are analyses performed for the unfactored and factored gravity inertia conditions to get demands from loads as shown in PSAR Table 3B-1. PSAR Figures 3E-2 and 3E-3 illustrate the magnitude of internal forces

(hoop and vertical membrane, moment and out-of-plane [OOP] shear) obtained for RB exterior wall and SCCV wall.

- Static and quasi-static analyses for loads using analysis methods described in PSAR Section 3.8.4.1.4 are performed using models described in PSAR Section 3B.4. Modeling requirements summarized in PSAR Table 3B-1 for the various static and quasi-static analysis cases are used. Responses (deformed shape) and demands from horizontal hydrostatic and hydrodynamic loads are illustrated for pool walls in PSAR Figures 3E-4(a) and 3E-4(b).
- Responses (deformed shape) and demands from containment internal pressure due to accident are shown in PSAR Figure 3E-5.
- Responses and demands from ground water and rock pressure loads applied are illustrated in PSAR Figures 3E-6 and 3E-7.
- Thermal analyses are performed by applying body loads to the shell and beam elements of the models described in PSAR Section 3B.5 using modeling requirements for thermal load cases (normal operating, design basis accident (DBA) presented in PSAR Table 3B-1. Deformed shapes for normal operating and DBA winter thermal conditions are illustrated in PSAR Figure 3E-8. Internal forces from DBA with summer exterior temperature is shown PSAR Figure 3E-9.
- Examination of the results of 1-g static SSI, static and quasi-static and thermal analyses performed in the PSAR Appendix 3E figures indicate they are reasonable.

Based on the above review of PSAR Appendix 3E, the staff finds that the applicant has demonstrated adequate implementation of the 1-g static SSI, static and quasi-static and thermal analyses performed for the applicable loads (other than dynamic seismic) of the IRB structures to determine design structural demands (other than dynamic seismic) consistent with the design and analysis methodology/procedures in PSAR Section 3.8.4.1.4 using appropriate FE models described in PSAR Appendix 3B. The staff also finds that the results of the preliminary 1-g static SSI, static and quasi-static, and thermal analyses performed are reasonable.

3.14.5.4 3E.4 Conclusion

Based on the above review of PSAR Appendix 3E, as amended, the staff concludes that the applicant has demonstrated adequate implementation of the 1-g static SSI, static and quasi-static and thermal analyses performed for the applicable loads (other than seismic) of the IRB structures to determine design structural demands (other than dynamic seismic) because it is consistent with the design and analysis methodology/procedures in PSAR Sections 3.8.4.1.4. The staff also concludes that the preliminary 1-g static SSI, static and quasi-static, and thermal analyses results are reasonable and adequate to perform the preliminary design evaluation of the IRB structures pursuant to the regulatory requirements of 10 CFR 50.34(a)(4) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35. The results of these preliminary analyses and design evaluations are reported for selected IRB structures in PSAR Appendices 3F through 3H, the staff evaluation of which are provided in the corresponding sections of this report.

3.14.6 3F DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT

3.14.6.1 3F.1 Introduction

PSAR Appendix 3F, submitted by CPA Supplement 1 dated October 1, 2025 (TVA 2025-TN13050), focuses on the preliminary analysis and design evaluation results of the SCCV portion of the containment and provides estimates of the available margins for the structure. The PSAR states that design demands used in the evaluations are obtained from the structural analyses discussed in PSAR Appendices 3C and 3E performed using the one-step approach and the FE models discussed in PSAR Appendix 3B. The design demands are obtained from design load combinations that assess the operational and accident structural demands on the structure.

The applicant amended PSAR Appendix 3F by CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029) and CPA Supplement 7 dated March 2, 2026 (TVA 2026-TN13043).

3.14.6.2 3F.2 Regulatory Evaluation

10 CFR 50.34(a)(4) (TN249) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures of the facility including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

PSAR Appendix 3F documents implementation of the design criteria and design analysis procedures described in PSAR Section 3.8.1 and resulting margins for the SCCV portion of containment that will be constructed of DP-SC modules which is a novel feature. The staff previously reviewed the SCCV design criteria and procedures primarily against the guidance in Section 6.0 of NRC-approved LTR NEDC-33926P-A and found it acceptable (subject to CP conditions L-CP-3.8-1 and L-CP-3.8-2) in Section 3.8.1.3.1 of this report. Therefore, the staff review of PSAR Appendix F is against the design criteria and analysis procedures for the SCCV in PSAR Section 3.8.1.3 and Section 6.0 of NRC-approved LTR NEDC-33926P-A.

3.14.6.3 3F.3 Technical Evaluation

The staff reviewed PSAR Appendix 3F, as amended, against the design criteria and analysis procedures for the SCCV in PSAR Section 3.8.1, and noted the following:

- Design of the SCCV portion of containment is implemented in accordance with Section 6.0 of NRC-approved LTR NEDC-33926P-A, which is consistent with the design criteria and procedures in PSAR Sections 3.8.1.2 and 3.8.1.4. The design parameters are shown in PSAR Table 3F-1 and compliant with LTR NEDC-33926P-A.
- Design evaluations of SCCV portion of containment are performed based on design demands obtained from structural analyses and FE models discussed in Appendices 3B, 3C and 3E for the load combinations in PSAR Table 3.8-1 assessing structural demands conservatively on an element-by-element basis (without averaging). This is consistent with or allowed by the criteria in PSAR Sections 3.8.1.3. and 3.8.1.4, noting that averaging the structural demand results across adjacent elements is also allowed by LTR NEDC-33926P-A.
- Design capacities for different limit states used to calculate reported demand-to-capacity ratios (DCRs) are based on material properties and geometric configuration of each section

and computed using the design rules for DP-SC SCCV in Section 6.0 of NRC-approved LTR NEDC-33926P-A, which is consistent with the design acceptance criteria in PSAR Section 3.8.1.5.

- The results provide capacities for out-of-plane (OOP) shear strength, combined OOP shear forces interaction, and allowable stress capacities for steel plates and concrete infill. DCRs and interaction checks are reported for the critical components namely SCCV wall, top slab and inner mat foundation.
- Enveloped DCRs are used to produce contour plots for each design criteria and also are used to identify critical locations and vulnerabilities for specific limit states.
- Enveloped DCRs and interaction checks for the SCCV wall, top slab and inner mat foundation are shown in the PSAR Figures 3F-1 through 3F-12 plots, for critical criteria including concrete principal compressive stress, faceplate Von Mises yield stress, OOP shear, and OOP shear interaction on diaphragm plates.
- Review of DCRs in PSAR Figures 3F-1, 3F-2, 3F-5, 3F-6, 3F-9, and 3F-10 indicate the SCCV concrete principal compressive stress DCR are generally well below the 0.5–0.625 range for each of the wall, top slab and base mat. The steel faceplates von mises stress DCRs are generally well below 0.75, but in the 0.75–1.0 range near the top of the SCCV wall with some expected localized exceedances around the main steam line penetrations because of the discontinuity around these penetrations. The preliminary evaluation results thus indicate good margins for the concrete principal stresses and faceplate von mises stresses for the SCCV.
- Review of PSAR Figures 3F-3, 3F-4, 3F-7, 3F-8, 3F-11, and 3F-12 indicate the OOP shear plots show locally high DCRs in the 0.875–1.0 range and related design exceedances where the RB slabs and wing walls intersect the SCCV wall as expected because of support interface. The PSAR explains that, however, some of the elements with design exceedances are within the thickness of the intersecting component or are outside the critical section for shear, which is away from the support interface. Also, the FE mesh is approximately half the SCCV wall thickness which would allow element averaging across four elements for future evaluation. Similar observations can be made for the SCCV top slab and inner basemat which show shear-critical behavior.
- Additional evaluations performed or design updates developed to address the exceedances will be summarized in the FSAR.
- Two types of connections are used to connect various IRB SC components as illustrated in PSAR Figure 3F-13 as allowed by section N9.4 of ANSI/AISC N690-18 referenced in Section 6.14 of LTR NEDC-33926P-A: (1) full-strength connections and (2) over-strength connections. A full-strength rigid T-connection as shown in PSAR Figure 3F-13 is appropriately used to connect the SCCV wall to the base mat foundation.

Based on the above review, the staff finds that the applicant has adequately documented implementation of the preliminary design evaluation of the SCCV portion (using DP-SC modules) of containment in accordance with the design criteria and evaluation procedures in PSAR Section 3.8.1 which is based on LTR NEDC-33926P-A. The staff also finds that the results are reasonable and margins assessed for concrete principal stress and steel faceplate von mises stress (which incorporate in-plane membrane, in-plane shear and OOP flexure demands) generally meet or exceed the acceptance criteria and therefore acceptable. The staff also finds that the observed design exceedances for OOP shear and OOP shear interaction limit state in the SCCV components and possible resolution path reasonably explained. The applicant stated that additional evaluations performed or design updates developed to address the exceedances will be included in the FSAR, which the staff finds acceptable to review during the operating license application stage. The staff thus finds that the applicant has provided adequate information of the preliminary design evaluation of the SCCV portion of containment (using DP-SC modules) pursuant to 10 CFR 50.34(a)(4) for issuance of a construction permit.

The staff further noted during its review that, while PSAR Appendix 3F provided preliminary design evaluation and results for the SCCV portion of containment using DP-SC modules, it did not include a preliminary analysis and design evaluation results for the Class MC components of the containment pressure-resisting boundary (e.g., the closure head, air locks, penetrations) the design criteria and design analysis procedures for which is addressed in PSAR Section 3.8.2. The staff notes that the functions of the Class MC portions of the containment pressure boundary include mitigating the consequences of accidents. In this regard, the staff noted that Section 3F.1 of Rev 1 of PSAR Appendix 3F, as amended by CPA Supplement dated March 2, 2026 (TVA 2026-TN13043), states that the major Class MC components are represented in the IRB FE model in PSAR Appendix 3B to ensure accurate loading demand values can be derived from the structural analyses summarized in PSAR Appendices 3C through 3H. The revised PSAR further states that design analysis of the Class MC components will be performed per methodology in PSAR Section 3.8.2 using demands from the FE models, by the fabricator during the procurement process. The preliminary loads on MC components have been reviewed by the applicant and determined reasonable to continue with the current design and a summary of the results and margins for those components will be provided during future licensing activities.

Based on the above, the staff finds that there is reasonable assurance that the final design of Class MC portions of containment pressure boundary will be implemented conforming to the design bases and criteria in PSAR Section 3.8.2 because the major Class MC components are included in the FE element models described in PSAR Appendices 3B and 3C and demands under design loads and load combinations can be accurately derived from these models for design implementation by the fabricator. Because design implementation of the CRN-1 containment Class MC components is in progress, the staff further finds it reasonable to defer its review of the design implementation and results of the containment Class MC components to the FSAR stage because the design of Class MC containments have been successfully implemented in the past in accordance with the ASME BPVC Code, Section III, Subsection NE, as supplemented by RG 1.57, in several operating LWRs.

3.14.6.4 3F.4 Conclusion

Based on the review of PSAR (Rev 1) Appendix 3F, and as amended by CPA Supplement 5 and CPA Supplement 7 dated March 2, 2026 (TVA 2026-TN13043), the staff concludes that the applicant has adequately documented implementation of the preliminary design evaluation of the SCCV portion (using DP-SC modules) of containment because they are in accordance with the design criteria and evaluation procedures in PSAR Section 3.8.1 and based on NRC-approved LTR NEDC-33926P-A. The staff also concludes that the results are reasonable and margins assessed for concrete principal stress and steel faceplate von mises stress are acceptable because they generally meet or exceed the acceptance criteria. The staff also concludes that it is reasonable for additional evaluations performed or design updates developed to address the observed design exceedances for the OOP shear and interaction limit state in the preliminary evaluation to be included in the FSAR and reviewed by staff during the operating license application stage because the PSAR included a reasonable explanation and feasible potential resolution. The staff also concludes that it is reasonable to defer its review of the design implementation and results of the containment Class MC components to the FSAR stage because the major containment Class MC components are included in the 3-D FE models for determination of design demands and the design of Class MC containments have been successfully implemented in the past in accordance with the ASME BPVC, Section III, Subsection NE, as supplemented by RG 1.57, in several operating LWRs. The staff thus finds that the applicant has provided adequate information of the preliminary analysis and design evaluation of the SCCV portions (which is a novel feature) and Class MC portions of the CRN-1 containment pursuant to the regulatory requirements of10 CFR 50.34(a)(4) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.14.7 3G DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT INTERNAL STRUCTURES

3.14.7.1 3G.1 Introduction

PSAR Appendix 3G, submitted by CPA Supplement 1 dated October 1, 2025 (TVA 2025-TN13050), focuses on the preliminary analysis and design evaluation results of the Diaphragm Plate Steel-Plate Composite (DP-SC) Reactor Pressure Vessel (RPV) pedestal and provides estimates of the available margins for the structure. The PSAR states the design demands used are obtained from structural analyses for design load combinations that assess the operational and accident structural demands on the structure. The PSAR also states that design evaluations for other containment internal structures are not provided for preliminary evaluation as these structures do not contribute to the global response of the integrated RB and their design is still progressing.

The applicant amended PSAR Appendix 3G by CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029).

3.14.7.2 3G.2 Regulatory Evaluation

10 CFR 50.34(a)(4) (TN249) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures of the facility including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

PSAR Appendix G documents implementation of the design criteria and design analysis procedures for the containment internal structures in PSAR Section 3.8.3 and resulting margins focused on the RPV pedestal that will be constructed of DP-SC modules novel feature. The staff previously reviewed the RPV design criteria against SRP Section 3.8.3.II and found it acceptable in Section 3.8.3.3 of this report. Therefore, the staff review of PSAR Appendix G is against the design criteria and analysis procedures for the RPV pedestal in PSAR Section 3.8.3.

3.14.7.3 3G.3 Technical Evaluation

The staff reviewed PSAR Appendix 3G, as amended by CPA Supplement 5, against the design criteria and analysis procedures for the RPV pedestal in PSAR Section 3.8.3, and noted the following:

- Preliminary design implementation and results focused on the RPV pedestal is acceptable for CPA since it is the most safety-significant containment internal structure (CIS) and involves the DP-SC novel design feature.

- Design of the RPV pedestal is implemented in accordance with Section 5.0 of NRC-approved LTR NEDC-33926P-A, consistent with the design criteria in PSAR Section 3.8.3.2. The design parameters are shown in PSAR Table 3G-1 and compliant with LTR NEDC-33926P-A.
- Design evaluations of RPV pedestal performed are based on design demands obtained from structural analyses and FE models discussed in Appendices 3B, 3C and 3E for the load combinations in PSAR Table 3.8-7 that assess the operational and accident structural demands, which is consistent with the criteria in PSAR Section 3.8.3.3. and 3.8.3.4.
- Design capacities for different limit states used to calculate reported demand-to-capacity ratios (DCRs) are based on Section 5.0 of NRC-approved LTR NEDC-33926P-A, consistent with the design acceptance criteria in PSAR Section 3.8.3.5.
- Enveloped DCRs for the two notional half checks (involving in-plane membrane, in-plane shear and out-of-plane [OOP] moments) reported in PSAR Figures 3G-1 and 3G-2 indicate maximum values of approximately 0.75 near the base of the RPV pedestal and lower in other locations. Acceptance criteria for DCRs are less than or equal to 1.0.
- Enveloped DCRs for OOP shear and OOP shear interaction reported in PSAR Figures 3G-3 and 3G-4 are in the 0.875–1.0 range at the top or bottom at interfaces with the CEPSS or basemat with localized exceedances are reasonable, with lower DCRs elsewhere. The exceedances observed at the inner mat foundation all lie within the thickness of the SCCV mat foundation and will be further addressed in the FSAR.
- A full-strength rigid T-connection as shown in PSAR Figure 3F-13 is appropriately used to connect the RPV pedestal to the base mat foundation.

Based on the above review, the staff finds that the applicant has documented an adequate demonstration of implementation of the preliminary design of the DP-SC RPV pedestal in accordance with the design criteria and evaluation procedures in PSAR Section 3.8.3. The staff also finds that the results are reasonable and margins provided in the RPV pedestal preliminary design generally meet or exceed the acceptance criteria, with some localized exceptions, and thus acceptable. Additional evaluations or design updates to address localized exceptions within the SCCV basemat foundation area will be addressed in the FSAR.

3.14.7.4 3G.4 Conclusion

Based on the review of PSAR Appendix 3G, as amended, the staff concludes that the applicant has documented an adequate demonstration of implementation of the preliminary design of the DP-SC RPV pedestal (the most critical containment internal structure [CIS]) because it is consistent with the design criteria and evaluation procedures in PSAR Section 3.8.3.and the assessed margins in the preliminary RPV pedestal design generally meet or exceed the acceptance criteria. The staff also concludes it is reasonable for additional evaluations or design updates to address localized design exceedances to be provided and reviewed at the FSAR stage. The staff thus concludes that the applicant has adequately demonstrated implementation of the preliminary design evaluation of the RPV pedestal pursuant to the regulatory requirements of 10 CFR 50.34(a)(4) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.14.8 3H DESIGN DETAILS AND EVALUATION RESULTS FOR THE REACTOR BUILDING STRUCTURE

3.14.8.1 3H.1 Introduction

PSAR Appendix 3H, submitted by CPA Supplement 1 dated October 1, 2025 (TVA 2025-TN13050), evaluates the design and provides estimates of the available margins for key structural members of the Reactor Building (RB) of DP-SC or SC construction. The design demands are obtained from structural analyses of design load combinations which assess the operational and accident structural demands on the structure.

The applicant also amended the PSAR Appendix 3H consistent with CPA Supplement 5 dated January 7, 2026 (TVA 2026-TN13029).

3.14.8.2 3H.2 Regulatory Evaluation

10 CFR 50.34(a)(4) (TN249) requires the applicant to provide information of a preliminary analysis and evaluation of the design and performance of structures of the facility including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

PSAR Appendix H documents implementation of the design criteria and design analysis procedures described for the RB in PSAR Section 3.8.4.1 and resulting margins for key Reactor Building structural members that will be constructed of DP-SC modules which is a novel feature or conventional SC. The staff previously reviewed the Reactor Building design criteria and procedures primarily against the guidance in Section 5.0 of NRC-approved LTR NEDC-33926P-A for DP-SC and ANSI/AISC N690-18, Appendix N9 (as endorsed in RG 1.243) and found it acceptable (subject to CP conditions L-CP-3.8-1 and L-CP-3.8-2) in Section 3.8.4.3.1 of this report. Therefore, the staff review of PSAR Appendix H is against the design criteria and analysis procedures for the Reactor Building (RB) in PSAR Section 3.8.4.1 and Section 5.0 of NRC-approved LTR NEDC-33926P-A for DP-SC and ANSI/AISC N690-18, Appendix N9 (as endorsed in RG 1.243) for conventional SC components.

3.14.8.3 3H.3 Technical Evaluation

The staff reviewed PSAR Appendix 3H, as amended, against the design criteria and analysis procedures for the SCCV in PSAR Section 3.8.4.1, and noted the following:

- Design of the RB structure using DP-SC modules is implemented in accordance with Section 5.0 of NRC-approved LTR NEDC-33926P-A and Reactor Building conventional SC components (wing walls) in accordance with ANSI/AISC N690-18, Appendix N9. The design guidance in these documents is consistent with the design criteria in PSAR Section 3.8.4.1.2.
- The design parameters (member thickness, faceplate thickness, yield strength, concrete compressive strength, and diaphragm plate/tie rod spacing) for RB DP-SC/SC structural components are shown in PSAR Table 3H-1 and are compliant with LTR NEDC-33926P-A for DP-SC and ANSI/AISC N690-18, Appendix N9 for SC.
- Design evaluations of the Reactor Building (RB) structure are performed based on design demands obtained from structural analyses and FE models discussed in Appendices 3B, 3C and 3E for the load combinations in PSAR Table 3.8-7 assessing structural demands

conservatively on an element-by-element basis (without averaging). This is consistent with or allowed by the criteria in PSAR Sections 3.8.4.1.3. and 3.8.4.1.4, noting that averaging the structural demand results across adjacent elements is also allowed by LTR NEDC-33926P-A and ANSI/AISC N690-18.

- Design capacities for different limit states used to calculate reported demand-to-capacity ratios (DCRs are based on material properties and geometric configuration of each section and computed using the design rules for DP-SC RB in Section 5.0 of NRC-approved LTR NEDC-33926P-A, and for SC in Appendix N9 of ANSI/AISC N690-18. These are consistent with the design acceptance criteria in PSAR Section 3.8.4.1.5.
- The results provide capacities for uniaxial tensile strength, compressive strength, out-of-plane (OOP) flexural strength, in-plane shear strength, OOP shear strength, as well as combined capacities considering OOP shear forces interaction, and in-plane membrane forces and OOP moments interaction.
- Envelope DCRs and interaction checks from the preliminary evaluation are reported in terms of notional half 1, notional half 2, OOP shear and OOP interaction for the critical Reactor Building (RB) components (i.e., Reactor Building [RB] below-grade exterior wall, above-grade exterior wall, outer mat foundation, wing walls, subgrade floor (EL-14.8 m) and refuel floor).
- Enveloped DCRs (maximum element DCR over all load combinations) are used to produce contour plots for each design criteria for the critical components and shown in PSAR Figures 3H-1 through 3H-6.
- Review of PSAR Figures 3H-1 and 3H-2 for RB exterior wall show acceptable DCRs with locally high notional half-DCRs where subgrade slabs intersect with the wall. Some notional half-exceedances are observed near grade where the thick grade slab intersects the exterior wall around the steam tunnel attributed to high thermal and pressure demands in the steam tunnel and pool under accident conditions.
- Review of PSAR Figures 3H-3 for the outer mat foundation, the notional half DCRs are quite low, with high OOP shear DCR near the intersection with the walls above due to discontinuity similar to the inner mat foundation. For the wing walls, PSAR Figure 3H-4 show general DCR values in the 0.5–0.75 range with some locally high OOP shear DCRs near the top where they are attached to thick slabs near grade.
- Review of PSAR Figures 3H-5 and 3H-6 indicate that all DCRs for the subgrade floor are low. The refueling floor has generally acceptable DCRs with some local exceedances around the south side where the thick fuel pool walls below intersect which can be attributed to discontinuity (reentrant corner).
- Additional evaluations performed or design updates developed to address the exceedances will be summarized in the FSAR.
- Connections are designed per methodology described in PSAR Section 3F.5, consistent with LTR NEDC-33926P-A and Section N9.4 of ANSI/AISC N690-18, and examples illustrated in PSAR Figure 3F-13. The Reactor Building (RB) exterior wall-to-outer mat foundation and RB reinforced concrete roof-to-RB exterior wall are full-strength rigid L-connections. Wing walls, and slab-at grade-to-RB and SCCV walls are connected using full-strength rigid T-connections, while below-grade slabs are connected to the RB and SCCV walls using overstrength rigid T-connections.

Based on the above review, the staff finds that the applicant has documented an adequate demonstration of implementation of the preliminary design evaluation of the Reactor Building (RB) key structural components of DP-SC or SC construction in accordance with the design criteria and evaluation procedures in PSAR Section 3.8.41 which is based on NRC-approved LTR NEDC-33926P-A or ANSI/AISC N690-18 (as endorsed in RG 1.243). The staff also finds that the reported results are reasonable and margins assessed generally meet or exceed the acceptance criteria and therefore acceptable. The applicant stated that additional evaluations performed or design updates developed to address some localized design exceedances will be included in the FSAR, which the staff finds acceptable to review during the operating license application stage. The staff thus finds that the applicant has provided adequate information of the preliminary design evaluation of the key RB structural components pursuant to the requirements of 10 CFR 50.34(a)(4) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.14.8.4 3H.4 Conclusion

Based on the above review of PSAR Appendix H, as amended, the staff finds that the applicant has documented an adequate demonstration of implementation of the preliminary design evaluation of the Reactor Building (RB) key structural components of DP-SC or SC construction because it is consistent with the design criteria and evaluation procedures in PSAR Section 3.8.41 based on NRC-approved LTR NEDC-33926P-A or ANSI/AISC N690-18 (as endorsed in RG 1.243). The staff also concludes that the reported results are reasonable and acceptable because the margins assessed generally meet or exceed the acceptance criteria, and localized exceedances observed will be addressed further in the FSAR. The staff thus concludes that the applicant has provided adequate information of the preliminary design evaluation of the key Reactor Building (RB) structural components pursuant to the requirements of 10 CFR 50.34(a)(4) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.14.9 3I COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF SEISMIC CATEGORY STRUCTURES

3.14.9.1 3I.1 Introduction

PSAR Appendix 3I describes the major computer programs used in the design and analysis of the BWRX-300 seismic category structures.

The applicant amended PSAR Appendix 3I by CPA Supplement 6 dated February 4, 2026 (TVA 2026-TN13031).

3.14.9.2 3I.2 Regulatory Evaluation

- 10 CFR 50 (TN249), Appendix A, GDC 1–*Quality standards and records*, requires, in part, that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. A quality assurance (QA) program shall be implemented for design and appropriate records of the design shall be maintained.
- 10 CFR 50.34(a), "Preliminary Safety Analysis Report," sub-paragraph (3)(ii), requires applicants to address the design bases (in this case the computer programs used in the design and analysis of Seismic Category I structures) and the relation of the design bases to

the principal design criteria, in this case how quality standards required by GDC 1 are met for computer programs used in design and analysis of seismic category I structures.

- The acceptance criteria in SRP Section 3.8.1.II.4.F states the computer programs used in the design and analysis should be described and validated by any of the following procedures or criteria:
  i. The computer program is recognized in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration.
  ii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained by a similar and independently written and recognized program in the public domain. The test problems should be demonstrated to be similar to or within the range of applicability of the problems analyzed by the public domain computer program.
  iii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained from classical solutions or from accepted experimental tests or to analytical results published in technical literature. The test problems should be demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program.

3.14.9.3 3I.3 Technical Evaluation

The staff reviewed PSAR Appendix 3I, as amended by CPA Supplement 6 (TVA 2026-TN13031), on computer programs used for structural analysis and design of seismic Category I structures and noted the following:

- The PSAR, as amended, describes ACS SASSI v4, ANSYS v17, LS-DYNA R13.1.0 and SSI_StressCoordTrans01P as the computer programs used in the design and analysis of the BWRX-300 seismic category structures for CRN-1.
- The computer programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems.
- The computer codes used for design and safety analysis are qualified in accordance with the GVH) Quality Assurance Program Description (QAPD).

During the audit, the staff audited the following documents in the Electronic Reading Room (ERR).

- Ghiocel Predictive Technologies Inc, ACS SASSI NQA Version 4.3.4 V4 License, Quality Assurance Certificate of Conformance Issued to GE Hitachi, June 24, 2022
- Design Basis Record DBR-0057949 Rev 2 GEVH SASSI04P, Acquisition Acceptance Test Report, PLM Part 006N4770, Revision 2, October 2025, Proprietary
- GEH DBR-0066734, Revision 1, Testing for Engineering Software SASSI04P-4.3.4-S September 2025, Proprietary
- ANSYS Release 17.2 Version: Windows 10 Intel x86 platform Certificate of Conformance provided by ANSYS Inc. to GE-Hitachi Energy Americas LLC, September 21, 2016
- GEH/GNF DBR-0025738, Revision 1, ANSYS Version 17.2 Level 2 – Mechanical (Objective: to determine suitability of ANSYS v17.2 as a Level 2 computer code by running a

suite of test cases, provided by ANSYS (Structural Mechanics Verification Testing Package Version 17.0), and comparing the results to expected values)

- GEH DBR-0073007, Revision 0, Fire Rating Resistance of BWRX-300 Diaphragm Plates Steel Composite (DP-SC) System (13.1 Appendices - Appendix A: Benchmarking Problems to Validate the Advanced Analysis Method for Fire Resistance Rating Calculations using LS-DYNA Nonlinear Solvers (Problem #1 to #4)

The staff notes that SASSI, ANSYS and LS-DYNA are programs recognized in the public domain and have been used by applicants in previous licensing applications. The staff finds that PSAR Appendix 3I, as amended, lists and describes the computer programs primarily used in the analysis and design of seismic Category I structures as recommended in SRP Section 3.8.1.II.4.F. The NRC staff also finds that the verification methodologies discussed above for computer programs are consistent with SRP Section 3.8.1.II.4.F, and therefore acceptable. During the audit, the staff audited a sample of validation documents of computer programs ANSYS v17.2, SASSI v4.3.4, and LS-DYNA R13.1.0 primarily used for structural analysis and design for the CPA and confirmed that adequate verification and validation (V&V) documentation exists for these computer programs.

3.14.9.4 3I.4 Conclusion

Based on the above review, the staff concludes that PSAR Appendix I, as amended, adequately describes the computer programs used for the analysis and design of seismic category I structures. The NRC staff also concludes that the verification and validation documentation exists, and the methods discussed in PSAR Appendix 3I, as amended, for computer programs are consistent with SRP Section 3.8.1.II.4.F. Therefore, documentation used by the applicant is meets the regulatory requirements of 10 CFR 50.34(a)(3)(ii) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

3.14.10 3J COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF THE STRUCTURES, SYSTEMS AND COMPONENTS – MECHANICAL DESIGN

3.14.10.1 3J.1 Introduction

See Section 3.9.1 of this report.

3J.2 Regulatory Evaluation

N/A

3.14.10.2 3J.3 Technical Evaluation

See Section 3.9.1 of this report.

3.14.10.3 3J.4 Conclusion

See Section 3.9.1 of this report.

3.14.11 3K COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF THE STRUCTURES, SYSTEMS AND COMPONENTS – NUCLEAR FUELS

PRIME is an NRC-approved fuel rod thermal-mechanical computer code (see PSAR references 4.2-4, 4.2-5, 4.2-6, 4.2-7). CRN-1 BWRX-300 appears to be using PRIME within the scope of prior approvals, however such detailed consideration would appear in Section 4.2.
Appendix 3K.1.4 “TRACG v4:” the application of this thermal-hydraulic code to the CRN-1 BWRX-300 is reviewed in SE Section 15.11
Appendix 3K.1.1 “PANAC v11” and Appendix 3K.1.3 “TGBLA v6”: the application of these codes to CRN-1 BWRX-300 is reviewed in SE Section 4.3.

3.14.12 3L COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF THE STRUCTURES, SYSTEMS AND COMPONENTS – ENVIRONMENTAL AND RADIOLOGICAL

Appendix 3L.1.1 “GALE-BWR v3.2 is used to calculate the gaseous and liquid effluent releases from BWRs. As discussed in Chapter 11 of this SE, the applicant used the GALE-BWR methodology, consistent with the guidance in RG 1.112, Revision 1 (NRC 1974-TN13097) and NUREG-0016 to calculate annual average effluent releases from the BWRX-300 plant. The staff found the applicant’s approach to be reasonable and acceptable. The staff’s evaluation of the applicant’s approach for estimating effluent releases can be found in Sections 11.2 and 11.3 of this report.

3.14.13 3M COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF THE STRUCTURES, SYSTEMS AND COMPONENTS – SAFETY ANALYSES (PROBABILITY RISK ASSESSMENT AND DETERMINISTIC)

Sections 3M.1.1 through 3M.1.9 describe the suite of probabilistic risk analysis (PRA) computer codes and post-processing tools proposed for use. These tools are typical of those commonly used in industry PRA applications. As discussed in SE Section 15.5.6, the PRA has not been completed at CP stage. Accordingly, the staff has not evaluated the PRA related information as part of this safety review to support issuance of the CP. Therefore, no staff conclusions regarding the acceptability or use of PRA computer codes are required at this stage.

Appendix 3M.1.10 “TRACG v4:” the application of this thermal-hydraulic code to the CRN-1 BWRX-300 is reviewed in SE Section 15.11.

Appendix 3M.1.11 “GOTHIC v8:” the GOTHIC computer methodology for measuring containment response is provided in NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” Revision 3 (PSAR Reference 6.2-2). GOTHIC is also touched on in SE Section 15.5.

Appendix 3M.1.14, “PAVAN,” provides a brief description of the PAVAN code, which is used to estimate the dispersion characteristics for design-basis accidental releases to the EAB and LPZ. Section 2.3.4 of the CRN-1 ESP FSER provides additional discussion and details on this code.

Appendix 3M.1.15, “ARCON,” provides a brief description of the ARCON code, which is used to estimate the dispersion characteristics to the control room envelop.

3.14.14 3N AIRCRAFT IMPACT ASSESSMENT

3.14.14.1 3N.1 Introduction

The Aircraft Impact Assessment (AIA) is a design-specific evaluation of damage caused by the impact of a large commercial aircraft on the BWRX-300 plant. The evaluation includes potential structural damage, the effects of shock-induced vibration, and the effects of aviation fuel-fed fires. The analysis describes the design features and functional capabilities, with reduced use of operator actions, of the BWRX-300 and determines whether the following acceptance criteria in accordance with 10 CFR 50.150(a) (TN249) are met:

- The reactor core remains cooled, or the containment remains intact.
- Spent fuel cooling or fuel pool integrity is maintained.

3.14.14.2 3N.2 Regulatory Evaluation

As stated by the applicant, the AIA is performed in accordance with:

- 10 CFR 50.150(a)
- 10 CFR 50.150(b)
- 10 CFR 50.34(a)(13)

The applicant also follows the methodology and guidance for AIA in accordance with NEI 07-13, "Methodology for Performing Aircraft Impact Assessment for New Plant Designs," Nuclear Energy Institute, Revision 8, April 2011 as endorsed by RG 1.217, Revision 0, “Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts.”

The NRC staff reviewed the application for conformance with the above regulations and guidance in accordance with DNRL-ISG-2022-01, “Safety Review of Light-water Reactor Construction Permit Applications.”

SRP Section 19.5 provides guidance for meeting the requirements in 10 CFR 50.150(a)(1) and (b).

3.14.14.3 3N.3 Technical Evaluation

As stated in PSAR Appendix 3N, “Aircraft Impact Assessment,” of the CRNS CPA, the AIA was performed to assess the physical damage, shock-induced vibration damage, and fuel-fed fire damage using the methodology described in NEI 07-13 as endorsed by RG 1.127. The staff reviewed the applicant’s submittal related to the requirements of 10 CFR 50.150 to the point necessary to make a finding for the construction permit. The staff determined that the use of the RG methodology provides an acceptable approach to demonstrate compliance with 10 CFR 50.150. The RG incorporates realistic assumptions for aircraft characteristics, impact orientations, and subsequent structural and thermal effects. The applicant stated the key design features and functional capabilities to ensure that the BWRX-300 design can meet the acceptance criteria listed in Section 3N.1 following an impact of a large commercial aircraft include:

- Reactor Building, Containment, and Fuel Pool Structural Integrity
- Reactor Shutdown, Containment Isolation, and Core Cooling
- Containment, Core, and Fuel Pool Heat Removal Capability
- Fire Barriers and Fire Protection Features

The applicant stated that the structural design of the Reactor Building (RB) and protection of penetrations are adequate to prevent perforation and structural collapse of the RB. The staff did not review the analysis performed but acknowledges that the description in the CPA is sufficient to make a finding that the CP is acceptable. The staff will further review the analysis during the future OL application. Key design features identified for Reactor Building, Containment, and Fuel Pool Structural integrity identified by the applicant are:

- Reactor Building structural design
- individual Reactor Building penetration designs
- design of the Reactor Building polar crane

The applicant also stated that reactor core and spent fuel pool cooling system, structure, and components (SSCs) are protected from physical, shock/vibration, and fuel-fed fire damage because the integrated RB structure will not be perforated by an aircraft impact and physical barriers are utilized to ensure jet fuel does not enter the Reactor Building penetrations.

Design features to assure reactor shutdown, containment isolation, and core cooling are identified by the applicant as:

- design of control systems that permit operators to manually initiate reactor SCRAM, containment isolation, and isolation condenser cooling
- SSCs credited to implement these actions, including:
  - The Control Rod Drive (CRD) system
  - main steam, condensate, and feedwater containment isolation valves used to isolate the containment
  - The Isolation Condenser System (ICS)
  - The Reactor Isolation Valves (RIVs) used to isolate the reactor

Design features to assure containment, core, and fuel pool heat removal identified by the applicant are:

- passive containment cooling system design and layout
- ICS design and layout
- fuel pool design and depth of water maintained above spent fuel

Key design features noted by the applicant to implement fire protection measures are:

- design and fire rating of walls and roof of the Reactor Building as determined by the Fire Hazards Analysis
- physical barriers used to ensure jet fuel does not enter the integrated RB penetrations

The staff recognizes that the construction permit application does not include the final design information typically available at the OL stage. Therefore, the staff's review is focused on the information in the PSAR necessary to make a finding for issuance of a CP. By implementing the endorsed NEI 07-13 process, the applicant has identified the key design features and functional capabilities necessary to maintain the required safety functions, which is all the information needed at the CPA review stage. The staff finds that information provided by the applicant on key design features and functional capabilities is adequate to support issuance of a CP.

3.14.14.4 3N.4 Conclusion

The staff has reviewed the available information provided in the PSAR, and, for the reasons given above, concludes that the applicant's description of the BWRX-300 design-specific features provides adequate description of equipment necessary to ensure reactor core and spent fuel pool cooling in the integrated Reactor Building (RB) structure in the event of a large commercial aircraft impact at the current stage of review. Accordingly, the staff concludes that the applicants AIA, performed in accordance with NEI 07-13 meets the regulatory requirements for the construction permit review in that the applicant’s identification of key design features and analysis provides reasonable assurance that the facility will maintain core cooling, spent fuel pool cooling, and containment integrity following a large commercial aircraft impact with sufficient certainty for this stage of licensing. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets the regulatory requirements of 10 CFR 50.34(a) and other requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35.