15 SAFETY ANALYSES

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 15, "Safety Analyses," of the Tennessee Valley Authority's (hereinafter referred to as TVA's or the applicant's) Construction Permit Application (CPA) Preliminary Safety Analysis Report (PSAR). TVA submitted this CPA for a single unit, small modular reactor (SMR) plant at the Clinch River Nuclear (CRN) Site located in Oak Ridge, Roane County, Tennessee. The PSAR is based on the proposed construction of a 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 CRN-SAR, dated April 29, 2026 (ML26119A630, non-proprietary; ML26119A633, proprietary).

The staff reviewed Chapter 15 of the TVA CPA PSAR, as supplemented, using applicable regulatory requirements, regulatory guidance, and standards to assess the sufficiency of the preliminary information in the safety analyses for the issuance of a construction permit (CP) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." As part of this review, the staff reviewed Chapter 15 of the PSAR, with special attention given to design and operating characteristics, unusual or novel design features, and principal safety considerations. The staff also evaluated the PSAR to ensure the design criteria, design bases, and information related to construction are sufficient to provide reasonable assurance that the final design will conform to the design basis.

The staff has also considered technical and topical reports which supplement this chapter:

- NEDC-33934P, "BWRX-300 Safety Strategy," Revision 2, March 2026 (NRC 2026-TN13056, non-proprietary; ML26077A382, proprietary)
- NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," GE Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022 (GE Hitachi 2022-TN13095, non-proprietary; ML22168A013, proprietary)
- NEDC-34043, "BWRX-300 TRACG Application," Revision 1, May 2025 (GE Hitachi 2025-TN13070, non-proprietary; ML25141A241, proprietary)
- NEDC-34270P, Revision 1, "BWRX-300 Stability Analysis" (GE Verona 2026-TN13153, non-proprietary; ML26089A384, proprietary)
- NEDC-33910P-A, ""BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," Revision 2, June 2021 (GE Hitachi 2021-TN13156, non-proprietary; ML21183A261, proprietary)
- NEDC-33840P-A, "The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance," Revision 1, August 2017 (GNFA 2017-TN13157, non-proprietary; ML17230A011, proprietary)
- NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR II), Revision 31, November 2020 (Main: GNF 2020-TN13158, non-proprietary; ML20330A197, proprietary; Supplement: GNF 2020-TN13163, non-proprietary; ML20330A198, proprietary)

- NEDE-32176P, “TRACG Model Description,” GE Nuclear Energy, Revision 4, January 2008 (GE Hitachi 2008-TN13159, non-proprietary; ML080370276, proprietary)
- NEDE-32177P, “TRACG Qualification,” GE Nuclear Energy, Revision 3, August 2007 (GE 2007-TN13160, non-proprietary, ML072480083, proprietary)
- NEDE-32906P-A, “TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis,” GE Nuclear Energy, Revision 3, September 2006 (GE 2006-TN13161, non-proprietary; Part 1: ML062720300, proprietary, Part 2: ML062720309, proprietary)
- NEDE-33005P-A, Revision 2, “TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant Accident Analyses for BWR/2-6” (NRC 2018-TN13146)
- NEDE-33885P-A, “GNF CRDA Application Methodology” (NRC 2020-TN13147)
- NEDO-31960-A, “Licensing Topical Report, BWR Owners’ Group Long-Term Stability Solutions Licensing Methodology” (GE 1995-TN13154) and Supplement 1 (GE 1995-TN13155)

The staff’s reviews and evaluations for areas relevant to PSAR Chapter 15, including a summary of the application information reviewed, regulations and guidance used, and evaluation findings and conclusions, are discussed in the safety evaluation (SE) sections below for each specific review area.

15.1 Overview of the BWRX-300 Safety Analyses

15.1.1 Summary of Application

Section 15.1 of the PSAR presents a description of the foundational approach, objectives, and scope of the safety analyses performed for the BWRX-300 reactor design proposed for the CRN site. The safety analysis framework is stated to integrate the principles of defense-in-depth and risk-informed decision-making and leverage insights from operating experience, deterministic evaluation, and probabilistic assessment. The applicant states that this approach demonstrates the adequacy of design features and operational practices in mitigating postulated initiating events (PIEs) and associated hazards.

The safety analyses documented in Chapter 15 of the PSAR are intended to serve multiple regulatory and operational functions, including the following:

- Demonstrating compliance with the acceptance criteria established for each plant state, following a graded approach that applies more restrictive acceptance criteria to events of higher probability.
- Providing the analytical basis for the derivation and validation of plant technical specifications (TS) that govern normal operation, in accordance with the requirements of 10 CFR 50.34(a)(5) and supporting regulatory guidance such as NUREG-0800 and Regulatory Guide (RG) 1.70.
- Supporting the development of accident management procedures and guidelines, ensuring the plant’s resilience against a broad spectrum of operational transients and accident scenarios.

The scope of the safety analyses encompasses all relevant plant states: normal operation, anticipated operational occurrences (AOOs); design-basis accidents (DBAs); and design extension conditions (DECs), both with and without core damage. The BWRX-300 deterministic safety analyses (DSAs) utilize a layered approach consisting of baseline (BL-DSA), conservative (CN-DSA), and extended (EX-DSA) evaluations to systematically address PIEs and failures of mitigating functions, providing a comprehensive demonstration of plant safety across a range of scenarios.

The PSAR describes the event sequence development approach, which is further detailed in NEDC-33934P, herein after referred to as "Safety Strategy." The Safety Strategy establishes design rules for applying defense-in-depth through successive defense lines (DLs) and defines how hazards, failures, and PIEs are considered in the development of event sequences used in the safety analyses. In this framework, hazards or failures affecting plant structures, systems, and components (SSCs) may lead to PIEs that challenge the performance of fundamental safety functions. Event sequences are then developed by considering the PIE together with assumed successes or failures of the mitigation functions provided by the DLs. The Safety Strategy uses the frequency of either a PIE or an event sequence (consisting of the PIE with additional postulated failures of DLs) to categorize events as AOOs, DBAs, and DECs using frequency thresholds. Specifically, PSAR Section 15.2.2, "Categorization of Events According to Their Frequencies," defines the frequency range for each event category as the following:

- AOO (frequency greater than 1×10^{-2} per reactor-year)
- DBA (frequency between 1×10^{-2} and 1×10^{-5} per reactor-year)
- DEC (frequency less than 1×10^{-5} per reactor-year)

As described in the PSAR, the frequency used for event categorization may correspond either to the frequency of a PIE or to the frequency of an event sequence that includes the initiating event combined with the failure of one or more mitigating systems. PSAR Section 15.2.2 further states that the event categorization presented in the application is currently qualitative and that quantitative frequency estimates will be based on Level 1 Probabilistic Safety Assessment (PSA) results for the final safety analysis report (FSAR). As noted in PSAR Section 15.6.9, "Results of the Level 1 and Level 2 Probabilistic Safety Assessment," final PSA results will be provided in the FSAR.

15.1.2 Regulatory Evaluation

The staff based its review on the relevant requirements of the following regulations and guidance:

- 10 CFR Part 20 (TN283), "Standards for Protection Against Radiation."
- 10 CFR 50.2 (TN249), "Definitions."
- 10 CFR 50.34(a), "Contents of application, technical information – Preliminary safety analysis report."
- 10 CFR 50.35, "Issuance of construction permits."
- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."
- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

- BWRX-300 design specific principal design criteria (PDC) as required by 10 CFR 50.34(a)(3).
- General Design Criterion (GDC) 2, "Design bases for protection against natural phenomena," as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.
- GDC 4, "Environmental and dynamic effects design bases," as it relates to the requirement 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 accident (PA) conditions, including such effects as pipe whip and jet impingement.
- GDC 5, "Sharing of structures, systems, and components," as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.
- GDC 10, "Reactor design," as it relates to the reactor core and associated coolant, control, and protection systems being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including AOOs.
- GDC 13, "Instrumentation and control," as it relates to instrumentation and controls (I&C) provided to monitor variables over anticipated ranges for normal operations, for AOOs, and for accident conditions.
- GDC 15, "Reactor coolant system design," as it relates to the reactor coolant system (RCS) and associated auxiliary, control, and protection systems being designed with appropriate margin to ensure that the design conditions of the pressure boundary will not be exceeded during normal operations, including AOOs.
- PDC 17, "Electric power systems," as it relates to the electric power system(s) having sufficient capacity and capability to ensure that (1) the design limits for the fission product barriers are not exceeded as a result of the AOOs and (2) safety functions that rely on electric power are maintained in the event of DBAs. Further, each layer of defense shall have sufficient independence, redundancy, and testability to ensure the reliability of the functions being supported.
- PDC 19, "Control room," as it relates to the requirement that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and AOOs and to maintain it in a safe condition during DBAs.
- GDC 20, "Protection system functions," as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the plant does not exceed SAFDLs during any condition of normal operation, including AOOs.
- GDC 25, "Protection system requirements for reactivity control malfunctions," as it relates to the requirement that the reactor protection system be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods.
- PDC 26, "Reactivity control system redundancy and capability," as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded even during AOOs. This is accomplished by providing two independent reactivity control systems of different design principles. One system uses two diverse means of inserting

control rods, which ensures there is appropriate margin for malfunctions such as stuck rods. The second system shall be capable of inserting negative reactivity into the core to assure a reactor shutdown from full power operating conditions if both diverse means of inserting control rods fail.

- GDC 27, "Combined reactivity control systems capability," and PDC 28, "Reactivity limits," as they relate to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.
- PDC 29, "Protection against anticipated operational occurrences," as it relates to the design of the protection and reactivity control systems to assure (1) an extremely high probability of accomplishing fundamental safety functions in the event of AOOs and (2) a high probability that SAFDLs are not exceeded as a result of AOOs.
- GDC 31, "Fracture prevention of reactor coolant pressure boundary," as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.
- GDC 34, "Residual heat removal," as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded.
- GDC 35, "Emergency core cooling," as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.
- GDC 55, "Reactor coolant pressure boundary penetrating containment," as it relates to the isolation requirements of small-diameter lines connected to the primary system.
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (NRC 2021-TN8013).
- Regulatory Guide (RG) 1.70 (NRC 1978-TN12810), Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition."
- Interim Staff Guidance (ISG) DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications."

15.1.3 Technical Evaluation

The applicant's evaluation of its nuclear power plant design includes analyses of the plant's responses to postulated equipment failures or malfunctions. Such analyses help to determine the limiting conditions for operation, limiting safety system settings, and design specifications for components and systems important to safety necessary to protect public health and safety. The staff reviewed Chapter 15 of the TVA CPA PSAR, Revision 1, for conformance with applicable regulatory requirements above. In performing this review, the staff referred to regulatory guidance listed above and associated standards to assess the sufficiency of the preliminary safety analyses for the BWRX-300 design for the issuance of a CP in accordance with 10 CFR Part 50 (TN249).

In addition to the information in the PSAR, the staff's evaluation was also supported by a regulatory audit (NRC 2025-TN13059). As discussed in Section 1.1 of this report, the staff conclusions provided in this section are based on information provided by the applicant in the PSAR, supplemental docketed information, and information reviewed as part of the regulatory audit.

The applicant used a new approach for the design and analysis of the BWRX-300 that primarily focuses on defense-in-depth concepts. The Safety Strategy describes the design and safety assessment framework used in the development of the BWRX-300 design. In general, the applicant described their design using terminology and concepts derived from the Safety Strategy as an alternative to the framework defined in 10 CFR Part 50 and its supporting guidance. Although the Safety Strategy provides detailed information related to design philosophy and the DL approach adopted for the BWRX-300, the staff has not yet accepted the Safety Strategy approach as a means to meet applicable regulatory requirements. Therefore, staff does not endorse or make a determination on the acceptability of the BWRX-300 Safety Strategy process as part of this safety evaluation. Instead, the staff evaluated the design and analysis that resulted from the engineering design process utilized by the applicant to develop the BWRX-300 and makes findings relative to applicable regulatory requirements contained in 10 CFR Part 50. However, because the PSAR was written using the Safety Strategy framework, this SER Chapter describes how Safety Strategy concepts may align with applicable regulatory requirements. Approval or acceptance of the Safety Strategy will be determined through a separate action, such as review of a topical report.

At the time of CRN-1 CPA review, construction activities for a BWRX-300 reactor were underway at the Darlington Nuclear Power Plant site in Bowmanville, Ontario, Canada. Consequently, during its regulatory audit (NRC 2025-TN13059), the staff observed several underlying engineering calculations and supporting documents applicable to the BWRX-300 standard plant that were labeled as design-basis records for Ontario Power Generation (OPG) DNNP-1 project, or the BWRX-300 OPG Conceptual Design Phase. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary), the applicant stated the plant designated as CRN-1 in the TVA PSAR is the BWRX-300 standard plant design as referenced in documents related to the Darlington plant, along with site-specific considerations where applicable (e.g., siting, external events), and TVA has not specified any design deviations from this BWRX-300 standard plant. Based on this response, the staff determined the underlying engineering documents are applicable to the BWRX-300 design proposed for the CRN-1 site.

Consistent with the requirements of 10 CFR 50.34(a)(4), the applicant provided a preliminary analysis and evaluation of the design and performance of SSCs in PSAR Chapter 15. The staff notes that the guidance supporting implementation of the regulations in 10 CFR 50.34(a) for the PSAR and 10 CFR 50.34(b) for the FSAR, generally assumes that similar analysis methods are used in both the PSAR and the FSAR. Therefore, updates to the safety analysis between the PSAR and FSAR licensing phases would focus on pertinent information developed since submittal of the PSAR rather than significant changes to analysis methods. As discussed in the sections below, the staff has identified areas where the PSAR analysis methods, while acceptable for issuance of a construction permit, would need to be enhanced to support issuance of an operating license. For example, analyses supporting the FSAR should be performed using evaluation models and analytical methodologies that include appropriate figures of merit, acceptance criteria, and cover the range of permitted fluctuations in the operating domain for the BWRX-300 design. Therefore, the staff has highlighted areas in this report section where additional information related to analysis methods will be needed at the OL phase.

When evaluating the information provided in the PSAR and docketed supplements, the staff considered if the applicant adequately described the proposed design, including principal architectural and engineering design criteria, and major features needed for protection of the health and safety of the public. Specifically, the staff considered if sufficient information was available to authorize construction of SSCs described in the PSAR (e.g., preliminary SSC description, including capacities, capabilities, and operating conditions); if further technical or design information required to complete the safety analysis will be supplied in the FSAR; and if the facility constructed as described in the PSAR could reasonably be licensed at the OL phase, assuming additional technical information were provided with the operating license application. The staff also considered if issues not fully addressed at the CP phase could be addressed at the OL phase through adjustments to the operational envelope or minor design modifications, rather than significant design or construction changes (e.g., through use of appropriate technical specifications or operating limits rather than redesign of a major SSC). Therefore, in these cases, consistent with 10 CFR 50.35(a), the staff concluded that the PSAR and supplemental information was sufficient to support issuance of the CP and the additional technical information can be provided in the FSAR.

15.1.3.1 Categorization of Events

Consistent with the BWRX-300 Safety Strategy framework, the PSAR defines event sequences by postulating failures of DL functions. For example, beginning with an AOO, the PSAR evaluates progression to a DBA by assuming failure of DL2 mitigation functions. Progression to a DEC is evaluated by assuming the DBA combined with a failure of the DL3 mitigation functions. The staff notes that the applicant refers to these DL failures as “common cause failures” (CCFs); however, this terminology differs from the use of the term CCF in probabilistic risk assessment (PRA), where it refers to failure of two or more components in a system due to a single shared cause (NUREG-2122 (NRC 2013-TN13307), RG 1.200 (NRC 2020-TN7806). The PRA is discussed in PSAR Section 15.6, “Probabilistic Safety Assessment.” In contrast to the PRA term, as stated in Section 2.2.3, “Foundational D-In-D Analysis Concepts,” of the Safety Strategy, these DL failures are implemented as assumed failures of the associated digital technology platform, such that the functions within that DL are assumed not to respond to changing plant conditions (i.e., protective features associated with the DL are considered “frozen”). The staff finds that this approach provides a reasonable framework for evaluating the effectiveness of successive DLs in mitigating PIEs. However, a DL approach that credits non-safety-related SSCs or a determination of design-basis event (DBE) categorization on the assumption of an event sequence frequency does not directly align with the regulatory terminology contained in 10 CFR Part 50, including Appendix A and 10 CFR 50.2. Therefore, the staff performed a detailed evaluation of the PSAR Chapter 15 event categorization process to understand how the Safety Strategy concepts may align with regulatory requirements. The staff also evaluated whether the applicant considered a sufficiently broad spectrum of transient and accident events in its safety analysis and categorized each event to be consistent with regulatory requirements and consistent with guidance contained in RG 1.70 (NRC 1978-TN12810), SRP Chapter 15, and DNRL-ISG-2022-01, or other acceptable approaches. Detailed evaluation of categorization of events is provided in the following subsections.

<u>Postulated Initiating Events Identification</u>

PSAR Section 15.1.2, “Identification of Failures and Hazards,” describes the process used to systematically identify SSC functional and human failures that initiate a PIE. Specifically, the applicant states that it used plant level failure analysis: (1) functional failure analysis (FFA) and (2) human failure event analysis (HFEA). PSAR Section 15.1.2.1 states that the FFA identifies plant systems or equipment with potential to challenge a fundamental safety function by using failure modes and effect analyses of the plant systems.

PSAR Section 15.1.2.2, “Human Failure Event Analysis,” describes the HFEA used to identify those human failure events that could potentially initiate abnormal or accident event sequences. If a single human failure event results in the loss of capability of multiple components, the resulting PIE includes the effects of these failures. Although some human failure events will result in the same effects as PIEs identified by the FFA, the HFEA is expected to identify additional, unique human failures not included in the FFA. However, as stated in PSAR Section 15.1.2.2, the inputs needed to complete the HFEA are incomplete at the PSAR stage. Consequently, the HFEA will be completed prior to submittal of the FSAR.

Based on the information provided in the PSAR, the staff finds that the preliminary analyses, including the FFA, provide sufficient information to demonstrate that the applicant has identified the range of potential initiating events and meets the requirements in 10 CFR 50.34(a) and supports issuance of a CP in accordance with 10 CFR 50.35. Completion of the HFEA and further quantification of human failure-related PIEs can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

Frequency Thresholds

Once PIEs are identified, the applicant categorizes each PIE into various event categories based on their frequency using qualitative and quantitative approaches. The staff reviewed the proposed frequency thresholds for delineation between AOOs, DBAs, and DECs described in PSAR Section 15.2.2. The staff also notes there are several exceptions to the frequency-based classification that are described in the Safety Strategy report.

The staff finds the 1×10^{-2} per reactor year AOO-DBA threshold to be acceptable because it is consistent with the definition of AOO contained in 10 CFR Part 50, Appendix A. Specifically an event with a frequency of 1×10^{-2} per reactor year or greater would be expected to occur one or more times during a 100 year period. This corresponds to the 10 CFR Part 50, Appendix A, definition for AOOs as “conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit.”

The staff agrees that the use of the 1×10^{-5} per reactor year DBA-DEC threshold is a reasonable demarcation between DBA and DEC. While NRC regulations and SRP Section 15.0 do not define a quantitative threshold, this value is consistent with NRC precedents for beyond-design-basis events, such as anticipated transient without scram (ATWS) treatment in NUREG-1780 (NRC 2003-TN13181), *Regulatory Effectiveness of the Anticipated Transient Without Scram Rule*, and aligns with the examples of postulated accidents and beyond-design-basis event in SRP Chapter 15. Additionally, this demarcation between DBA and DEC is applied with deterministic exceptions described in the Safety Strategy, Section 2.2.3. The staff notes the specific inclusion of certain LOCA events as described in the Safety Strategy within the DBA category regardless of frequency. The staff reviewed the DBAs included in the PSAR and determined that all postulated accidents required by 10 CFR Part 50 were included in the DBA category, with the exception of the rod drop accident (as discussed in SER Sections 4.6, “Design of Reactivity Control Systems,” and 15.5.5, “Analysis of Design Extension Conditions Without Core Damage”) and certain LOCA locations upstream of the reactor pressure vessel (RPV) isolation valve assemblies (as discussed in PSAR Section 15.2.4.6.4, “Bounding Scenarios for DEC Pipe Breaks”). In addition, rod withdrawal events mitigated by Safety Class 1 (SC1) SSCs, initially proposed as DEC based on sequence frequency in the PSAR, will be recategorized in the FSAR (SER Section 15.5.5).

PSAR Section 15.2.2 states that event categorization presented in the CPA is qualitative and that quantitative frequency estimates will be provided in the FSAR. When categorizing events based on frequency, the Safety Strategy proposes consideration of uncertainties for DBA and DEC event sequences. The staff notes that uncertainties exist across the full spectrum of event frequencies and are inherent in PSA-based analyses. Guidance such as NUREG-1855 (NRC 2017-TN7798), Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," discusses the consideration of uncertainties in risk-informed decision making. While sequences near the AOO/DBA threshold can generally be categorized with higher confidence based on empirical evidence, a broader consideration of uncertainties across all frequency ranges could further strengthen confidence in accurate event categorization and the resulting SSC and function classification. When developing the quantitative frequency estimates, the applicant should consider uncertainties in the frequency estimates, as informed by applicable operating experience, for all PIEs and event sequences categorized according to frequency.

When the design and safety analysis is finalized in the FSAR, the staff will review the implementation of the frequency thresholds for AOOs, DBAs, and DEC in the FSAR to verify that PIE categorization process appropriately reflects the qualitative and quantitative criteria, including consideration of uncertainties.

Categorization Based on PIE vs. Event Sequence Frequencies

PSAR Section 15.2.2 states that event sequence frequencies, rather than the frequency of the PIE, are utilized to determine the event category. PSAR Section 15.2.2 states:

> "An event sequence consists of a PIE, an assumed failure of a mitigating function(s), and the DL function success that mitigates the PIE. The event sequence category is based on the sequence frequency, not the frequency of the PIE which initiates the sequence. The event category assigned to an event sequence may be different than the event category assigned to the PIE that initiated the sequence because the event sequence may include additional failures that make the sequence less likely to occur."

This is an alternate way to categorize DBEs than described in existing regulatory guidance (e.g., RG 1.70 (NRC 1978-TN12810), SRP Section 15.0), which treats the PIE as a discrete event that does not include subsequent SSC failures or successes. The staff evaluated the event sequence binning approach and considered how it aligns to the relevant requirements in 10 CFR Part 50, as described below.

In accordance with 10 CFR 50.34(a)(4), applicants for a CP are required to develop preliminary analysis and evaluation addressing the design and performance of SSCs, with the objective of assessing risk to public health and safety and margins of safety during normal operations, transients, and accidents anticipated during the life of the facility. The selection of DBEs defines the scope of the preliminary analysis and evaluation of SSC design and performance. The categorization of DBEs in this framework is based on the expected frequency of the initiating event consistent with the safety-related definition in 10 CFR 50.2, specific regulations such as 10 CFR 50.46 for LOCAs, and Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants." Consistent with risk concepts, graded performance criteria are established based on initiating event frequency. For example, more frequent DBEs (i.e., AOOs) need to meet more restrictive performance criteria compared to less frequent DBEs (postulated accidents). Specifically, for AOOs, the design must be capable of meeting the SAFDL performance criterion specified in the GDCs. **[[**

]]. However, if an SSC is needed to demonstrate that the performance criterion in the 10 CFR 50.2 "safety-related" definition is met during and following a DBE, then the SSC must be classified as safety-related. Since an AOO is a DBE, **[**

]. This concept is further explained in a letter dated July 31, 2025, "U.S. Nuclear Regulatory Commission Staff Response to Inquiry to Support BWRX-300 Safety Strategy Licensing Topical Report Review," (ML24255A092 [proprietary], NRC 2025-TN13170 [non-proprietary]) and the detailed staff evaluation of the applicant's alternate approach to rely on non-safety-related SSCs to mitigate AOOs is provided in Section 15.1.3.5 of this report.

An approach to event categorization has been described and implemented in NRC guidance documents, such as RG 1.70 (NRC 1978-TN12810) and SRP Chapter 15. SRP Section 15.0, "Introduction – Transient and Accident Analysis," Revision 3, specifies an approach that includes the following elements:

- Design-basis events are categorized as either AOOs or postulated accidents based on initiating event frequency, consistent with 10 CFR Part 50.
- An AOO should not generate a postulated accident without other faults occurring independently.
- AOO acceptance criteria should limit damage to allow resumption of operation.
- Only safety-related SSCs are used to mitigate AOOs and postulated accidents.

The approach described in SRP Chapter 15.0 assumes that safety-related SSCs will be used to mitigate AOOs to the more restrictive SAFDL acceptance limits. This approach provides assurance that failure of non-safety-related SSCs during an AOO will not result in the event progressing to a postulated accident. Therefore, the staff has deemed that the SRP Chapter 15 approach meets the requirements of 10 CFR Part 50. **[[**

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Although PSAR Table 1.9-20, "Conformance with Regulatory Guides," indicates that the CPA conforms with RG 1.70, and PSAR Table 1.9-15, "Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)," indicates conformance with SRP Chapter 15.0 criterion related to AOO acceptance criteria and general conformance with other SRP Chapter 15 sections, the applicant has used an alternate approach to mitigate AOOs. Consistent with 10 CFR Part 50, and as described in the Safety Strategy report, the applicant credits non-safety-related SSCs supporting DL2 functions to mitigate AOOs to the SAFDL acceptance criteria. SSCs supporting DL2 functions are classified as Safety Class 3 (SC3) (e.g., see PSAR Table 3A-1, "Preliminary BWRX-300 Component Classification List"). **[[**

]]. The applicant also considers failure of the DL2 mitigation function following an AOO, but considers this combination as an event sequence. PSAR Section 15.2.2 states that categorization of events is based on the event sequence frequency. Further, the event sequence consists of a PIE in combination with assumed failures of mitigation functions and the DL success that mitigates the PIE. Consequently, the event sequence frequency of an AOO followed by a DL2 failure is determined by multiplying the AOO initiating event frequency by the failure probability of the DL2 mitigation function. The resulting event sequence frequency is lower than the AOO initiating event frequency, which then can move the AOO initiated sequence into the DBA category.

The staff determined that the approach to categorize AOO initiated DBEs based on event sequence frequency rather than initiating event frequency can be an alternative to the approach described in RG 1.70 and SRP Chapter 15. The applicant's event sequence approach can meet the requirements of 10 CFR Part 50 provided two types of analyses are performed for each AOO initiating event: (1) an analysis where the non-safety-related SSCs supporting DL2 functions are assumed to perform their credited safety function, **[[**

]], and (2) an analysis where the same AOO initiating event is analyzed and demonstrates that the safety-related SC1 SSCs supporting DL3 functions are sufficient for **[[** **]]** (herein referred to as Type 1 and Type 2 analyses, respectively). The Type 1 analysis credits DL2 mitigation of the AOO initiating event, while the Type 2 analysis considers only DL3 mitigation of the AOO in conjunction with failure of DL2 functions (which defines an "event sequence" as described in the PSAR). The specific analysis acceptance criteria for the AOO and accident analysis are evaluated further below.

PSAR Section 15.2, "Identification, Categorization and Grouping of Postulated Initiating Events and Accident Scenarios," states the fault evaluation process used for PIE identification and selection produces results that are generally consistent with the events identified as AOOs and DBAs in SRP Chapter 15.0, "Introduction – Transient and Accident Analysis," and complies to the regulations referenced therein. While the PIEs may be consistent with events identified in SRP Chapter 15.0, use of the event sequence frequency for determining event categorization and treatment can systematically bias the categorization process toward less frequent event categories. Therefore, in the course of reviewing PSAR Chapter 15, the staff focused on ensuring that initiating events were analyzed against the appropriate acceptance criteria regardless of their assigned event sequence frequency. Based on its review, the staff agrees that the Chapter 15 safety analysis correctly categorizes AOOs PIEs as AOOs and PSAR Section 15.5.3, "Analysis of Anticipated Operational Occurrences," provides the corresponding analysis of these transients. However, the staff found that when the applicant analyzed the same AOO PIE but only credited safety-related SC1 SSCs (AOO Type 2 analysis described above that assumes failure of the non-safety-related DL2 mitigating functions), the applicant redefined the same event as a postulated accident for the purposes of the PSAR analysis. The staff notes that such an event would still be considered an AOO under 10 CFR Part 50 requirements, **[**

]]. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary), the applicant confirmed that the events analyzed in PSAR Section 15.5.4, "Analysis of Design Basis Accidents," provide a representation of the same AOO PIEs assumed in PSAR Section 15.5.3 but with the assumption that only safety-related SC1 SSCs are available for mitigation. The letter also provided clarification by mapping each Type 1 AOO analysis with its Type 2 AOO analysis. The staff finds that the supplemental information provided in the letter and the information in the PSAR is sufficient and meets the regulatory requirements of 10 CFR 50.34(a) and supports issuance of a CP in accordance with 10 CFR 50.35. Further categorization and demonstration of AOOs can be reasonably left for later consideration and will be reviewed with the FSAR at the OL licensing phase. The applicant must ensure the AOO analysis contained in the FSAR addresses all AOO initiating events and demonstrates that **[[**

]].

Additionally, in the letter (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary), the applicant states that "NEDC-33934P provides the necessary demonstration that the BWRX-300 Safety Strategy framework meets the regulatory criteria NRC staff specified in its letter dated July 31, 2025." The staff clarifies that NEDC-33934P only describes a general framework and that detailed design information or necessary analysis of the design is still required to demonstrate the BWRX-300 design compliance with applicable regulatory requirements. Therefore, the applicant needs to demonstrate that the BWRX-300 design meets NRC regulatory criteria in the safety analysis report submitted pursuant to 10 CFR 50.34, with an individual licensing application. Section 15.1.3.6 of this SER includes the staff's evaluation of the applicant's proposal to credit non-safety-related SSCs in the design-basis transient analyses of the BWRX-300.

15.1.3.2 Categorization of Event Types

The staff evaluated whether the events were categorized by type such that the analysis of AOOs and postulated accidents, within a type, encompasses a variety of cases, each designed to produce effects or results that challenge designated safety limits. In this context, all events within a specific event group have the same initial effect on the reactor plant (e.g., decrease in coolant temperature).

PSAR Section 15.2.3, "Grouping of Events According to Type," states that the PIEs (faults) are grouped according to the resultant change in plant parameters into the following categories:

- temperature decrease (TD) events—decrease in core coolant temperature
- pressure increase (PI) events—increase in reactor pressure
- reactivity increase (RI) events—reactivity and power distribution anomalies
- inventory increase (II) events—increase in reactor coolant inventory
- inventory reduction (IR) events—decrease in reactor coolant inventory
- non-reactor events—these events are non-core related such as fuel handling accidents

The staff evaluated the groupings above and finds that while they are not identical to the categories identified in SRP 15.0, Subsection I.1.B, they are consistent with expected groupings for a passive boiling water reactor (BWR). Staff notes the absence of the decrease in RCS flow rate type of AOOs and concludes that this is reasonable considering the BWRX-300 is a natural recirculation BWR design (i.e., there are no recirculation pumps to drive core flow). While control rod motion, feedwater temperature changes, and turbine load changes all lead to core thermal power changes, which leads to a change in RCS flow rate, the event types are defined based on the initiating effect that leads to a transient or postulated accident. Because BWRX-300 doesn't have recirculation pumps, there would not be an event type that has an RCS flow change as an initiating event. However, events that may impact RCS flow are addressed under other event type groups.

The staff considered the possible case variations of AOOs and postulated accidents presented in the PSAR to verify that the applicant has identified the limiting cases. PSAR Section 15.2.4," Postulated Initiating Events and Accident Scenarios," states bounding events are selected in each fault group for each event category and for the applicable DSA layer (e.g., baseline, conservative, and extended), and the resulting events selected are listed in Table 15.2-2, "Bounding Events Transient (Non-LOCA) and LOCA," and analyzed in Section 15.5, "Deterministic Safety Analyses." The staff evaluated the applicant's identification of limiting or nonlimiting AOOs and postulated accidents with particular attention to the bases used for comparison. The specific staff evaluation of each event type is contained in Section 15.5 of this report.

15.1.3.3 Use of Reliability Targets

Under the Safety Strategy framework, the frequency of an event sequence that assumes failure of credited DL mitigation functions is calculated as the frequency of the PIE multiplied by the target maximum failure probability (i.e., reliability target) assigned to the applicable DL functions.

In PSAR Section 15.2.2, the applicant specifies the following target reliability values:

- 1×10^{-2} failures per demand for DL2
- 1×10^{-4} failures per demand for DL3
- 1×10^{-3} failures per demand for DL4a

The applicant proposes to use a process where an AOO PIE (frequency greater than 1×10^{-2} per reactor-year) is combined with the assumed failure of DL2 mitigation results in a calculated sequence frequency in the DBA range (i.e., frequency between 1×10^{-2} and 1×10^{-5} per reactor-year). The applicant then evaluates the resultant AOO event sequence using DBA acceptance criteria. This involves performing a deterministic safety analysis to demonstrate mitigation by the next DL (safety-related DL3), assuming failure of the preceding one (non-safety-related, but potentially important to safety, DL2). In this manner, the reliability targets for DL2 and DL3 are used to establish numerical transitions between licensing event categories and to define the scope of deterministic safety analyses. See Sections 15.1.3.1 and 15.1.3.4 of this report for the staff's regulatory evaluation of this event categorization process and associated analysis acceptance criteria, respectively.

The staff recognizes that the applicant's approach provides a structured method for relating DL performance to the licensing basis event framework. The specified numerical reliability targets for DL2 and DL3 correspond to the numerical differences between the frequency ranges that delineate the AOO-to-DBA and DBA-to-DEC transitions, such that multiplication of the PIE frequency by the target value shifts the sequence into the next event category. In this respect, the targets provide a consistent mechanism for categorizing sequences and defining the associated analyses.

The staff notes that while alignment with these targets and with plant-level safety goals (e.g., core damage frequency [CDF] and large release frequency [LRF]) provides useful insight into overall plant performance, the staff would need to also review and evaluate the technical or risk-informed basis demonstrating that these values are appropriate for the specific SSCs or functions.

Therefore, staff clarifies that this review does not constitute approval of predefined numerical reliability targets for individual DLs, SSCs, or functions. Rather than endorsing specific reliability values, the NRC evaluates the plant design and the performance of SSCs to determine whether their capabilities, reliability, and functional performance are commensurate with their safety significance and consistent with applicable regulatory requirements, defense-in-depth principles, and overall plant safety performance objectives.

While sequence-based evaluations provide useful insights, the staff's assessment considers the integrated and cumulative contribution of SSCs across all applicable scenarios. Since individual SSCs or DL functions may be credited in multiple event sequences, their overall safety importance is evaluated in the aggregate. It is important to note that meeting overall plant-level safety goals alone does not ensure that the reliability of individual SSCs or functions is adequate, particularly when SSCs or functions mitigate multiple sequences; the contribution of each specific SSC/function to the overall safety case must be evaluated. Further, the impact of SSCs performing DL functions being out of service for testing or maintenance activities, dependencies among various DLs, and the impact of hazards (e.g., fire or external hazards) on SSC performance must also be considered.

Accordingly, the staff focuses on whether the individual SSCs/functions achieve appropriate safety performance, rather than relying solely on numerical reliability thresholds specified for DLs to transition between licensing event categories.

If the applicant uses numerical reliability targets for specifying SSC performance in the FSAR, the staff will review to ensure that the applicant includes the technical and risk-informed basis for the assumed SSC and function reliability and demonstrate that the reliability of individual SSCs or functions is appropriate when considering their individual role across all credited scenarios, rather than relying solely on DL-level targets. This basis should include how insights derived from the PRA were considered in the determination of reliability targets.

15.1.3.4 Analysis Acceptance Criteria

The staff evaluated whether the applicant has provided adequate and acceptable analysis acceptance criteria for each of the event categories.

PSAR Section 15.3, "Safety Objectives and Acceptance Criteria," identifies and describes the deterministic safety analysis (DSA) safety objective and acceptance criteria for AOOs and DBAs. PSAR Section 15.3 states,

> "[q]ualitative acceptance criteria are defined and met for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. Derived qualitative and quantitative acceptance criteria are used to analyze AOOs or DBAs. Qualitative acceptance criteria are supported by experimental data, prescribed by regulatory requirements, or prescribed by applicable codes and standards. The results of the quantitative safety analysis confirm the derived acceptance criteria."

The criteria for AOO initiating events mitigated with SSCs important to safety are provided in Table 15.3-1 "Anticipated Operational Occurrence Deterministic Safety Analysis Acceptance Criteria," and the criteria for AOO and DBA initiating events mitigated with safety-related SSCs are provided in Table 15.3-2, "Design Basis Accident Deterministic Safety Analysis Acceptance Criteria." PSAR Section 15.3 also states, "[t]he deterministic safety analyses DBA Event Sequences acceptance criteria are based on or derived from ensuring that the 10 CFR 50.46(b) acceptance criteria for emergency core cooling systems are met."

Fuel Acceptance Criteria

PSAR Table 15.3-2 states that the number of fuel rod failures is conservatively estimated for DBAs and provides the associated quantitative acceptance criteria that will be employed in the FSAR. PSAR Table 15.3-2 also provides “fuel cooling” acceptance criteria.

At a high level, the fuel acceptance criteria provided in PSAR Table 15.3-2 consider the following fuel failure modes:

- cladding embrittlement due to oxidation
- cladding ballooning and burst
- pellet-cladding mechanical interaction (PCMI)
- high temperature failure during reactivity-initiated accidents
- overheating of the fuel pellet (fuel melt)
- fuel fracturing due to DBA loading conditions

The staff reviewed the PSAR fuel acceptance criteria against the guidance contained in SRP Section 4.2, “Fuel System Design.” PSAR Table 1.9-4, “Conformance with NUREG-0800 (Chapter 4 Reactor),” indicates that the PSAR conforms with the SRP Section 4.2 criterion related to fuel failure and coolability. The applicant’s proposed failure criteria do not include a criterion for the onset of boiling transition (i.e., operation below the critical power ratio (CPR) for BWRs). SRP Section 4.2 identifies overheating of the cladding as a failure phenomenon and states that fuel cladding failure is presumed if local heat flux exceeds the CPR thermal design limit. The use of the CPR thermal design limit acceptance criteria has generally been used by the operating fleet for non-loss-of-coolant accidents (non-LOCAs). However, SRP Section 4.2 also acknowledges that CPR is not, itself, a failure mechanism; cladding can survive for some period of time post-boiling transition, but CPR is recommended until more mechanistic approaches are proposed. SRP Section 4.2 notes that alternate approaches to use of CPR should address cladding temperature, pressure, time duration, oxidation, and embrittlement. The applicant proposed to mechanistically address a suite of failure mechanisms, as listed above, instead of assuming failure occurs when the CPR is less than the minimum CPR (i.e., when boiling transition occurs). The NRC has not previously approved generic use of an alternate to the CPR cladding overheating failure criteria for all non-LOCAs. Consistent with the SRP Section 4.2 statement, fuel rods can survive some time in boiling transition, as has been demonstrated experimentally. For example, NUREG-0562, “Fuel Rod Failure as a Consequence of DNB or Dryout” (ML071780258), included departure from nucleate boiling (DNB) pressurized water reactor (PWR) and dryout (BWR) testing. The BWR dryout data reported in NUREG-0562 is from the Winfrith heavy water reactor, NRU Chalk River reactor, General Electric Test Reactor (GETR), and Halden reactor. Few rod failures were observed in the testing, with hundreds of rods tested in the cumulative NUREG-0562 database. Additionally, since the publication of NUREG-0562, additional BWR dryout tests were performed at the Power Burst Facility (PBF) and at Halden. No fuel failures were reported in the PBF OPTRAN 1-2 test series or the Halden IFA-613 test series, as reported in NUREG/CR-3948, “Experimental Results in the Operational Transient (OPTRAN) Tests 1-1 and 1-2 in the Power Burst Facility” (McCardell et al. 1985-TN13165), and Halden report HWR-666 (ML080350050), respectively. These test results show that rods that experience boiling transition are most likely to fail due to embrittlement because of cladding oxidation.

Since assuming failure at the onset of boiling transition is conservative, but not planned to be used by TVA, the OL applicant will need to identify and justify the post boiling transition true failure mechanisms and thresholds employed. The applicant proposes to address the failure mechanisms listed above, which the staff believes is reasonable for the CP stage to provide a reasonable demonstration of the radiological consequences of such events per 10 CFR 50.34(a)(1)(ii) and 10 CFR 50.34(a)(4), because the proposed criteria appear to address the failure modes that a rod would experience during a non-LOCA. The specific criteria proposed to address each failure mechanism are discussed below; however, the NRC will examine these in more detail during or before the review of the OLA. The NRC staff consider that the primary challenge with crediting post-boiling transition behavior will be methodology validation. The NRC staff notes that there are uncertainties in prediction of many of the important phenomena and in prototypicality of the available experimental data that could serve as the basis for allowing post-boiling transition survivability to be credited. These uncertainties would need to be addressed and that prototypicality of the data used to validate post-boiling transition models would need to be justified. Thermal-hydraulic and fuel thermal-mechanical models would also need to be validated. While there are post-boiling transition models in existing LOCA evaluation methodologies (EMs) and the TRACG computer code, these models will need to be demonstrated to be applicable to the different conditions of non-LOCAs. For example, non-LOCAs occur at much higher pressures than most LOCAs, and rewetting of the rod is much different in a non-LOCA than the rise of a quench front due to emergency core cooling system (ECCS) injection in the reflood phase of a LOCA. The NRC staff find that deferring the technical justification and validation for crediting of post boiling transition fuel survivability can be reasonably left for later consideration at the operating license (OL) phase because there is experimental evidence that rods can survive a limited time in dryout (e.g., see NUREG-0562 and NUREG/CR-3948). However, the ability to accurately predict the thermal-hydraulic and fuel thermal-mechanical behavior during non-LOCA dryout would need to be demonstrated with comparison to experimental data in the future. If the non-LOCA evaluation methodology cannot be comprehensively validated in the post-boiling transition regime, then the onset of boiling transition should be used as a failure metric, consistent with the rest of the U.S. fleet.

The NRC considers maintenance of cladding ductility to be vital to ensuring core coolability; therefore, the criteria used to preclude cladding embrittlement must not be exceeded. The criteria the applicant proposed to address cladding embrittlement due to oxidation were a peak cladding temperature (PCT) limit of 2200 degrees Fahrenheit (°F) and a transient oxidation limit as a function of pre-transient cladding hydrogen content. The transient oxidation limit proposed is consistent with Draft Regulatory Guide (DG)-1263, Revision 0, "Establishing Analytical Limits for Zirconium-Alloy Cladding Material" (NRC Undated-TN13303), Figure 2, and NUREG/CR-7219, "Cladding Behavior during Postulated Loss-of-Coolant Accidents" (NRC 2016-TN13166), Figure 24. The applicant also clarified, consistent with DG-1263, Revision 0, that for pre-transient cladding hydrogen content above 400 weight parts per million (wppm), a PCT limit of 2,050°F would be used in place of the 2,200°F PCT limit. The transient oxidation limit addresses the cladding embrittlement research finding of hydrogen-enhanced beta layer embrittlement. The lower PCT limit was established for cladding pre-transient hydrogen contents greater than 400 wppm due to a lack of experimental LOCA testing data on cladding with hydrogen contents above 400 wppm and temperatures above 2,050°F. The NRC staff finds the criteria to address cladding embrittlement due to oxidation to be reasonable for the issuance of a CP due to the fact that it meets the criteria for transient oxidation and PCT limits specified in DG-1263.

For cladding ballooning and burst, the applicant proposed to use a cladding rupture curve (rupture stress as a function of rupture temperature) consistent with the rupture curve in NEDE-33005P-A, Revision 2, "TRACG Application for Emergency Core Cooling Systems / Loss-of-

Coolant Accident Analyses for BWR/2-6" (NRC 2018-TN13146). Specifically, the applicant proposed to employ Figure 5.1-14 in NEDE-33005P-A, Revision 2. This rupture curve was previously only reviewed and approved for use in the NEDE-33005P-A, Revision 2, TRACG-LOCA evaluation methodology; it was not reviewed for application during non-LOCAs. The NRC staff finds the rupture curve reasonable for the issuance of the CP because it utilizes an NRC-approved rupture curve that may be applicable for non-LOCA scenarios. The NRC staff will need to evaluate the rupture curve to ensure that it is appropriate for the non-LOCA conditions for which it is proposed to be employed in the future.

To address high-temperature failure during reactivity-initiated accidents (i.e., failure due to high-temperature oxygen-induced embrittlement and fragmentation, and failure due to high-temperature cladding creep [rod ballooning and burst]), the applicant proposed to employ the NRC-approved high-temperature failure threshold that is defined by peak fuel enthalpy as a function of cladding differential pressure that is in Figure 3-8 of Global Nuclear Fuel's (GNF's) control rod drop accident (CRDA) method, NEDE-33885P-A, "GNF CRDA Application Methodology" (NRC 2020-TN13147). Since the approval of this failure threshold in the GNF CRDA methodology, the NRC staff issued RG 1.236 (NRC 2020-TN13149), "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents" (NRC 2020-TN13149), which provides an acceptable high-temperature failure threshold for control rod drop and control rod ejection accidents. GNF's failure threshold matches the RG 1.236 high-temperature failure threshold. The NRC staff finds the high-temperature failure threshold to be reasonable to predict high-temperature fuel rod failures during reactivity-initiated accidents because the threshold matches that previously approved in NEDE-33885P-A and that in RG 1.236.

To address PCMI failures, the applicant proposed to employ the PCMI failure threshold approved in the CRDA methodology, NEDE-33885P-A, Figure 3-7, that is expressed as peak enthalpy rise as a function of nodal peak pellet exposure. Similar to the high-temperature failure threshold, this threshold was approved in NEDE-33885P-A before the issuance of RG 1.236 (NRC 2020-TN13149), which contains PCMI failure curves. The PCMI failure curves in RG 1.236 are expressed as peak enthalpy rise as a function of excess cladding hydrogen content. As discussed in NEDE-33885P-A, the peak pellet exposure in the PCMI failure curve is a surrogate for cladding hydrogen content, and the NRC staff found the translation of cladding hydrogen content to peak pellet exposure for the purposes of the PCMI failure threshold to be acceptable. The peak enthalpy rise versus excess cladding hydrogen content failure threshold that was translated to peak enthalpy rise versus peak pellet exposure was based on the failure curves provided in a draft of RG 1.236 (i.e., DG-1327 [NRC 2016-TN13150]). While the PCMI curves changed from DG-1327 to RG 1.236, the RG 1.236 PCMI failure curve for temperatures less than 500°F and the NEDE-33885P-A Figure 3-6 failure curve that is translated into a function of peak pellet exposure are not significantly different. Additionally, RG 1.236 contains separate PCMI failure curves at or above 500°F and below 500°F, but the applicant proposes to only use one PCMI failure curve. TVA's proposal is consistent with NEDE-33885P-A and is based on the low-temperature PCMI failure curve in DG-1327 as opposed to the high-temperature failure curve. The low-temperature PCMI failure curve is more restrictive than the high-temperature PCMI failure curves, due to the fact that the cladding has less ductility at low temperatures, so the use of the one PCMI failure curve rather than two PCMI enthalpy rise failure curves is reasonable. The NRC staff finds the proposed PCMI failure threshold to be reasonable to calculate rod failure due to PCMI during rapid reactivity insertions because the threshold meets the guidance in previously approved NEDE-33885P-A and in RG 1.236.

Moreover, on PCMI, SRP Section 4.2 suggests that a transient cladding strain (TCS) limit should be employed to protect against PCMI. A cladding strain limit was proposed by the applicant, but based on the information presented, NRC staff could not conclude its adequacy during non-LOCAs or Type 2 AOOs. While the enthalpy rise PCMI failure threshold in the PSAR is expected to accurately predict PCMI failure during non-LOCAs where there is a rapid and large reactivity insertion, it is not clear if it would accurately address PCMI failures during non-LOCAs for which there is a slower power ramp rate. PCMI occurs due to the pellet expanding faster than the cladding, causing the cladding to fail due to brittle fracture. During a non-LOCA event that has a slower ramp rate compared to CRDAs, the pellet still expands due to the power increase like in a CRDA, but unlike rapid reactivity-initiated accidents, gaseous swelling of the pellet can occur due to the event time scale, allowing fission gas to diffuse to the grain boundaries. This can cause TCS to be larger for an event with a slower ramp rate than an event with a high ramp rate. As such, additional justification for a TCS limit when addressing PCMI during events that experience power increases is needed. Therefore, the NRC staff could not conclude that only an enthalpy rise PCMI failure threshold is appropriate and does not result in the number of fuel failures being underpredicted. Justification for a TCS failure threshold will be reviewed provided prior to or with the OL application. This technical information can reasonably be left for later consideration, consistent with 10 CFR 50.35, because this is not expected to significantly impact the radiological consequences of the events or the design if a TCS failure threshold is further justified at the OL stage.

The applicant proposed to assume failure when any fuel part of the fuel pellet, centerline or elsewhere, experiences temperatures greater than the melting temperature. This is consistent with SRP Section 4.2, and therefore, the NRC staff find that the PSAR reasonably addresses the fuel pellet overheating failure mode.

Evaluation of the analysis to ensure that there is no fuel fracturing due to DBA loading conditions is provided in Section 4.2 of this report.

Overall, the NRC staff finds that the fuel failure phenomena are reasonable and consistent with the requirements in 10 CFR 50.34(a) and supports issuance of a CP in accordance with 10 CFR 50.35 because they capture the relevant failure modes. As discussed above and consistent with 10 CFR 50.35, further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase to further justify the failure thresholds, including their acceptability over the range of conditions expected during non-LOCA events, for failure mechanisms that include the following: (1) cladding embrittlement due to oxidation, (2) cladding ballooning and burst, (3) PCMI, (4) high-temperature failure during reactivity-initiated accidents, and (5) fuel pellet overheating (fuel melt).

Additionally, the applicant will need to demonstrate the accuracy of the non-LOCA evaluation methodology in the post-boiling transition regime. The methodologies used to perform the non-LOCA analyses in the FSAR will need to be approved for use by the NRC staff.

<u>Reactor Coolant Pressure Boundary Acceptance Criteria</u>

The staff also reviewed the applicant's acceptance criteria related to protection of the reactor coolant pressure boundary (RCPB). The definition of safety-related SSCs in 10 CFR 50.2 states that these SSCs are relied upon to remain functional during and following DBEs to assure, among other things, "the integrity of the reactor coolant pressure boundary." SSCs relied upon to perform such functions are subject to special treatment requirements, including 10 CFR Part 50, Appendix B, and the quality assurance and design requirements of 10 CFR 50.55a, “Codes and standards,” which incorporates by reference the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC). In accordance with Section III of the ASME BPVC, maintaining RCS pressure within applicable ASME Service Level limits provides assurance of RCPB component integrity. Further, 10 CFR Part 50, Appendix A, GDC 15, “Reactor Coolant System Design,” states that the RCS be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. Consistent with these requirements, SRP Section 15.0 states that for AOOs, the pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressure, consistent with the limits established by the ASME BPVC for Service Level B conditions. Similar guidance is contained in SRP Section 5.2.2, “Overpressure Protection,” and the applicant states they conform with SRP Section 5.2.2 acceptance criteria in in PSAR Table 1.9-5, “Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems).”

As noted in PSAR Section 3.9.3.13, “Design of Pressure Relief Devices,” the RCS does not utilize safety or relief valves for overpressure protection. PSAR Section 3.1.2.6, “GDC 15 – Reactor Coolant System Design,” states that overpressure protection for AOO event sequences is provided in accordance with ASME BPVC, Section III, NB-7120(b) and (c) by Safety Category 3 functions (e.g., non-safety-related DL2) and the safety-related isolation condenser system (ICS). The applicant further states that this approach ensures that the overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition and that the calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition. The ASME BPVC generally describes Service Level B conditions as expected system pressure transient conditions associated with normal operation such as losses of feedwater or a loss of load. Service Level C conditions are unexpected system excess pressure transient conditions associated with unusual or abnormal system transients but still considered to be within the design basis, such as loss of load with a failure to scram. PSAR Chapter 3 (Table 3.9-1, “Design Condition Definition”) appropriately captures the Service Level conditions required by the ASME BPVC and identifies that ASME Service Level B applies to AOO PIEs. Further, PSAR Tables 15.3-1 and 15.3-2 provide RCPB acceptance criteria limits for AOOs of 110 percent of design pressure and for DBAs of 120 percent of design pressure.

As discussed in Section 15.1.3.1 of this report, the applicant proposes to determine event categories based on event sequence frequencies rather than PIE frequencies. Under this approach, the applicant classifies certain sequences initiated by AOO PIEs as DBAs when the non-safety-related DL2 functions are assumed to be failed. DBA acceptance criteria are then applied to this AOO-initiated event sequence (i.e., DBA pressure limits associated with Service Level C conditions) rather than AOO acceptance criteria. The staff acknowledges that AOOs followed by failure of DL2 mitigation functions should have a lower frequency than the AOO-initiating event. The staff will need complete design information, including PRA results and insights, reliability evaluations for SSCs performing DL2 functions, and final safety analysis results, at the OL licensing phase, to review and evaluate the proposed approach.

The preliminary transient and accident results of the BWRX-300 design presented in Tables 15.7-1, “Results Summary of AOO Events,” and 15.7-2, “Results Summary of the DBA and DEC Events – Non-LOCA,” all events, including DBAs scenarios, show that peak RCS pressures are maintained well below 110 percent of design pressure. The highest calculated AOO or DBA pressure presented in Tables 15.7-1 and 15.7-2 was approximately 1,300 pounds per square in absolute (psia), compared to the AOO limit of 1,650 pounds per square inch gauge (psig). Therefore, the staff has sufficient evidence to conclude that the preliminary BWRX-300 design

can satisfy applicable RCS pressure acceptance criteria for the AOO PIEs (Type 1 and Type 2 as denoted in Section 15.1.3.1 of this report) and meets the requirements in 10 CFR 50.34(a). Therefore, the preliminary analysis and results support issuance of a CP in accordance with 10 CFR 50.35 and further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.
The staff will review the DBA acceptance criteria at the OL phase to ensure an appropriate licensing basis is established prior to operation. Therefore, based on the above, the staff will defer its review of the RCPB acceptance criteria to the FSAR, in accordance with 10 CFR 50.35.

15.1.3.5 Plant Characteristics Considered in the Safety Analysis

10 CFR 50.34(a)(4) requires the preliminary safety analysis presented in the PSAR to include a determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility. RG 1.203 (NRC 2005-TN13304), “Transient and Accident Analysis Methods,” and SRP Section 15.0.2, “Review of Transient and Accident Analysis Method,” which the applicant states they conform with in PSAR Tables 1.9-20 and 1.9-15, provide guidance for developing and assessing evaluation models for accident and transient analyses. Consistent with the guidance in SRP Section 15.0, the staff evaluated whether the applicant appropriately considered the key plant parameters in the safety evaluation, including consideration of the permitted fluctuations within the operating band and uncertainties associated with reactor system parameters in order to identify appropriate initial conditions for transient and postulated accident analysis. The staff also evaluated whether the applicant ensured that the parameters and initial conditions used in the analyses are suitably conservative and the application discussed the bases (including the degree of conservatism) used to select the numerical values of the input parameters. Within this context, “suitably conservative” means the input values used for parameters and initial conditions represent either (1) an appropriately conservative set of possible conditions or (2) realistic conditions in concert with an appropriate treatment of the associated uncertainties and variabilities.

PSAR Table 15.5-3, “Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses,” provides a listing of the initial conditions assumed in the non-LOCA analyses. In reviewing this information, the staff notes that most assumptions for input parameters and initial conditions (e.g., power, pressure, flow, and temperature) use nominal values and in using nominal values, the results of the non-LOCA analyses in PSAR Section 15.5.3 show that there is not a challenge to the analysis acceptance criteria. Since the applicant’s analysis does not use initial conditions other than nominal, it is unclear to the staff whether appropriate biasing of the plant parameter assumptions to reflect normal operating fluctuations and uncertainties would result in significant reduction of margin. During the regulatory audit, the staff requested the applicant to perform sensitivity analyses using suitably conservative assumptions of one limiting AOO event and one limiting accident to assess the safety margins associated with these analyses. The staff requested the applicant to use either a previously approved uncertainty analysis methodology, or to follow a traditional bounding conservative approach, in order to demonstrate that variations in permitted operating conditions and uncertainties associated with reactor system parameters do not result in significant impact to the documented analysis results or conclusions.

In response to the staff’s request for additional sensitivity calculations, the applicant stated in a letter dated April 29, 2026 (ML26119A646), that the PSAR safety analysis was performed using its Safety Strategy approach. As discussed previously in this report, the staff has not yet approved or endorsed the BWRX-300 Safety Strategy approach and makes no determination at this time that following such an approach results in satisfying all applicable regulatory requirements. Specific to the request to perform a sensitivity analysis for a limiting AOO, the response states that the BL-DSA are performed using best-estimate methods and include conservatisms in certain plant parameter assumptions, such as setpoints, I&C timing, valve performance and scram performance. The staff notes that while these conservativisms are appropriate for a design-basis analysis, they are not sufficient to demonstrate with an appropriate level of confidence that AOO acceptance criteria are met. Furthermore, those parameters can only be considered to be design constraints, and the degree of conservatism cannot be quantified until the final design is available. Additionally, assuming nominal initial conditions, rather than accounting for expected fluctuations and uncertainties associated with key reactor system parameters permitted within the operating band may not provide reasonable assurance that AOO acceptance criteria are met. Such an approach would not fully demonstrate the design-basis safety capabilities of credited AOO mitigating SSCs (i.e., SC3 SSCs) consistent with the identified PDCs. The response further states the CN-DSA (which includes the requested postulated accident) accounts for uncertainties in initial conditions and methods. It explains that for the CN-DSA uncertainties are biased in the conservative direction. However, based on its review, the staff notes that most of the identified areas of biases and uncertainties are related to core and thermal-hydraulic parameters and do not fully address permitted fluctuations of key parameters within the operating band.

The applicant’s April 29, 2026, response states the BWRX-300 design’s margin of safety has been demonstrated by the defense-in-depth approach because of the use of multiple DLs, a layered DSA, and consideration of uncertainties for “the most important DL functions” (i.e., DL3, which are performed by SC1 SSCs). The staff acknowledges that the defense layer approach described by the applicant can provide additional defense in depth for certain DBEs, including AOOs that are primarily mitigated through use of DL2 functions and backed up by DL3 should the DL2 function fail. However, the use of a best-estimate baseline analysis for AOOs (BL-DSA) that does not account for expected variances of key parameters and initial conditions within the normal operating band does not reasonably demonstrate that AOO acceptance criteria (e.g., SAFDLs) can be met using SC3 SSCs. While acknowledging that the more conservative CN-DSA approach used for postulated accounts for more uncertainties than the AOO approach, the staff notes that this analysis also does not fully address expected variances of key parameters and initial conditions within the operating band. Due to these limitations, and the limitations in the stated capability of some SSCs performing DL3 functions, additional analysis is required, prior to issuing an operating license, using an approved methodology and a suitable set of input conditions and uncertainties that reflect expected operating variations to demonstrate that all applicable safety analysis acceptance criteria are met.

The staff’s evaluations of each of the AOO and DBA event analyses are in Sections 15.5.3 and 15.5.4 of this report. Based on its review of the PSAR, information reviewed during the regulatory audit, and supplemental information provided on the docket, the staff concludes that the analysis presented in Chapter 15 of the PSAR demonstrates that AOO and postulated accident non-LOCA scenarios can meet applicable acceptance criteria when assuming nominal operating conditions with limited consideration of uncertainties in key parameters and initial conditions. This provides a reasonable level of confidence that the preliminary analyses presented in the PSAR are consistent with the requirements in 10 CFR 50.34(a) and are sufficient to assess risk to public health and safety and adequacy of SSC capabilities. These preliminary analyses support issuance of a CP in accordance with 10 CFR 50.35. Further technical and design information needed to complete the safety analysis, performed using an approved methodology and a suitable set of input conditions and uncertainties that reflect expected operating variations, to demonstrate that all applicable safety analysis acceptance criteria are met, can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

The staff's review of key plant parameters and assumptions related to LOCAs, both design-basis and beyond-design-basis losses of coolant, is documented in Section 15.5.4 of this report.

SRP Section 15.0 also specifies that the range of values for plant parameters used in the PSAR Chapter 15 safety analysis is representative of fuel exposure or core reload. Further, the range of parameters should be sufficiently broad to cover the predicted fuel cycle ranges, to the extent practicable, based on the fuel design and acceptable analytical methodology. Based on a review of information provided in the PSAR, the staff concluded that the transient and accident analyses documented in the PSAR do consider fuel exposure for the non-LOCA events. Therefore, the staff concludes that, related to the consideration of a range of fuel exposure conditions for the non-LOCA events, the PSAR conforms to SRP Section 15.0 and is acceptable to meet the requirements of 10 CFR 50.34(a). Additional information on the staff's review of the analytical methodology can be found in Section 15.1.3.7 of this report.

PSAR Section 15.2.4 states, "the bounding event selection is performed for events that are initiated at full power conditions because they are expected to result in the most significant challenge to the fission product barriers." Based on information presented in the PSAR, the staff could not verify this statement and note that some events, such as reactivity insertion events, may be more limiting at off-rated power (e.g., low power) levels. In letters dated March 2, 2026 (TVA 2026-TN13052), and April 1, 2026 (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary), the applicant provided supplementation information and indicates that they intend the PSAR analyses to provide confidence in BWRX-300 design capabilities, whereas they intend FSAR analysis to set operating limits and identify the most limiting scenarios based on the final design. Staff recognizes that a systematic evaluation of initial conditions will be needed to support review of the OL application and eventual staff safety findings. Additionally, the applicant states that thermal limits for off-rated power conditions will be developed to ensure that these events are not more limiting with respect to event acceptance criteria.

To confirm that limiting initial conditions were appropriately identified, the staff requested that the applicant provide their evaluation of a limiting inadvertent control rod withdrawal initiated from low power conditions. While this type of event is described in PSAR Section 15.5.5.3 for initiation at full power conditions, the staff specifically requested analysis from startup or cold conditions. This request was made because reactivity-initiated events can be more severe when they initiate from lower power conditions. Such scenarios are typically considered in light-water reactor (LWR) licensing bases unless precluded by the design of the reactivity control system. Evaluation of this event is discussed in Section 15.5.5 of this report.

Consistent with SRP Chapter 15 and applicable regulatory requirements, the staff reviewed whether the applicant has specified only safety-related systems or components for mitigating AOO and postulated accident conditions, and whether the effects of single active failures in those systems and components were considered. Regarding the reliance solely on safety-related systems or components for mitigating AOO and postulated accident conditions, the applicant indicates in PSAR Table 1.9-15 that it does not conform to this approach (i.e., SRP 15.0, Criterion I.6) and instead proposes an alternate approach. The staff's evaluation of the proposed alternative is provided below. The staff's evaluation of the application of single failure requirements is provided in Section 15.1.3.6 of this report.

In September 2024, GVH submitted a letter requesting legal interpretation of the criteria defining "safety-related SSCs" in 10 CFR 50.2 to support its response to Safety Strategy audit questions (NRC 2024-TN13148). The NRC staff responded to GVH's inquiry in a letter dated July 31, 2025 (NRC 2025-TN13170). The NRC response letter explains that the SAFDLs have been used by applicants and licensees to support a demonstration of the capability of "safety-related" SSCs to maintain a safe shutdown condition during and following AOOs as described by 10 CFR 50.2, noting that the SAFDLs are not explicitly referenced in the 10 CFR 50.2 definition of "safety-related" SSCs, and the term is only used in the GDC. Further, the response letter explains that the NRC staff considers any SSC relied on to satisfy the GDCs, **[[**

]], to be considered an SSC that is "important to safety." This staff position is instituted in Generic Letter (GL) 84-01, "NRC Use of the Terms, 'Important to Safety' and 'Safety Related'" (NRC 1984-TN13167). GL 84-01 provides standard definitions for safety classification terms to clarify the broad term of "important to safety" and other classification terms which are considered synonymous with "safety-related," such as "safety-grade" or "safety system." See memoranda "Safety Classification Terminology – Proposed Standard Definitions" (NRC 1981-TN13168) and "Standard Definitions for Commonly-Used Safety Classification Terms" (NRC 1981-TN13169) for additional details on the use of these terms.

Considering the above, the NRC response letter reaffirms the standard definitions prescribed in the aforementioned memoranda, and those SSCs considered within the scope of "important to safety" and those SSCs within the scope of "safety-related." **[[**

]. As discussed in PSAR Section 15.2.1.1, the applicant relies on DL2 mitigation functions, such as the anticipatory trip system, to **[[**

]] an AOO. As noted in PSAR Table 3A-1, SSCs performing DL2 functions are categorized as SC3. Based on the above, the staff concludes that the applicant's proposed alternate approach to rely on non-safety-related SSCs to mitigate AOOs can be acceptable and align with the staff position communicated in its response letter (ML25175A199). As documented in Section 15.1.3.1 of this report, such an approach will require the applicant to ensure the AOO analysis contained in the FSAR addresses all AOO initiating events and demonstrates that **[[**

]].

Regarding appropriate quality standards for SSCs performing safety functions as described in NRC response letter (ML25175A199), PSAR Section 3.1.1.1 describes the applicant's approach to meeting the GDC 1 quality standards and states that the quality controls for risk-significant non-SC1 SSCs are consistent with the non-safety-related SSC quality controls described in SRP Section 17.5. The applicant further states that the criteria for determining risk significance of SSCs are included in PSAR Section 17.4.5, which describes quantitative risk significance criteria based on PRA results. The staff notes that final PSA results are not available at the CP licensing phase and use of only numeric risk-significance criteria may omit certain important to safety functions that are required by the GDCs and subject to GDC 1 quality controls. However, the staff considers identification of specific quality controls that will be applied to SSCs that support DL2 functions to be a matter that can be reasonably be left for later consideration consistent with 10 CFR 50.35.

15.1.3.6 Assumed Protection and Safety Systems Actions

<u>Identification of Credited SSCs and Demonstration of their Design Bases</u>
The staff evaluated PSAR Chapter 15 to verify that it lists the settings of all the protection and safety systems functions that are used (i.e., credited) in the safety evaluation and that the credited SSCs are consistent with their stated design bases for compliance with the general design criteria or design-specific PDCs as specified in PSAR Chapter 3. PSAR Table 15.5-5, "Defense Line Functions Used in Non-LOCA Analyses," identifies DL functions that are used in the non-LOCA analyses. In a letter dated March 2, 2026 (TVA 2026-TN13052), the applicant updated PSAR Table 15.5 to include cross-references to PSAR sections where descriptions of design features and associated requirements can be found and provided a high-level description of electrical power systems for certain DL2 functions, respectively. Based on its review of the PSAR and supplemental responses on the docket, it is unclear that the applicant has identified the entire complement of systems needed to support each credited function in the PSAR, primarily for the identified DL2 functions. For example, the response provided in the March 2, 2026, letter states that certain DL2 functions are "battery-backed"; however, the staff could not identify which of the three battery systems are supporting each of the DL functions. In addition, some of the credited DL2 functions rely on the use of turbine bypass system, and PSAR Section 10.4.4 states the turbine bypass valves are designed to fail closed upon loss of control power and then further characterizes the power loss from alternating current (AC) power sources like offsite or station auxiliary power. Therefore, it's unclear whether the credited DL function uses direct current (DC) power, AC power, or both, and which subsystem(s) support this function. The staff notes that SSCs supporting credited DL functions will need to be classified in accordance with the functional role they support in the AOO and postulated accident analysis. In light of the preliminary nature of the analysis required at the CP phase, and acknowledgement that the preliminary design will evolve during construction, the staff has determined that the applicant has provided sufficient information to meet the requirements of 10 CFR 50.34(a) for the CP phase. The applicant has identified the major features and components incorporated into the design for the protection of the health and safety of the public as required by 10 CFR 50.35. Therefore, this preliminary analysis supports issuance of a CP in accordance with 10 CFR 50.35 and further technical and design information needed for the finalization of SSC categorization based on the safety analysis can be reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

To understand how the applicant determines the DL functions that are credited in the PSAR, the staff audited the BWRX-300 Fault List, which, according to the PSAR, provides traceability between the plant design and the safety analysis. The DSA and PSA mature with the design and fault list is updated accordingly. Based on its regulatory audit (NRC 2025-TN13059) of the BWRX-300 Fault List that was used to complete the CPA, the staff observed a listing of credited DL1 requirements and DL functional assumptions that were not identified as design-basis requirements for individual systems in the supporting chapter of the PSAR. In a letter dated March 2, 2026 (TVA 2026-TN13052), the applicant provided a table of the DL1 requirements and DL functional assumptions for the events presented in PSAR Chapter 15. However, the response did not update the individual system descriptions in the PSAR to identify all the credited design-basis assumptions, and the response was only limited to the events included in PSAR Chapter 15. For example, the Chapter 15 AOO safety analysis assumes an instantaneous reduction of feedwater (FW) temperature of 50 degrees Kelvin (K; 90°F) based on a single feedwater heating failure (see PSAR Section 15.5.3.1.1); however, this is not identified as a design-basis requirement of the Condensate and Feedwater Heating System in PSAR Section 10.4.7, "Condensate and Feedwater Heating System." Acknowledging that the fault list will be updated once the design is final, the staff will review the FSAR to ensure that it demonstrates that credited DL functions are appropriately reflected in the associated SSC design.

In accordance with SRP Section 15.0 and 10 CFR 50.34(a)(3), the staff assessed if the Chapter 15 safety analysis includes the design bases and the relation of the design bases to the GDCs (or the PDCs, as applicable). The staff also assessed if credited SSCs can perform their intended safety functions, assuming consideration of single active failures and electrical power availability constraints when required by the GDCs (or the PDCs, as applicable). Specifically, the staff evaluated whether the Chapter 15 safety analysis demonstrates the protection and safety systems identified in PSAR Section 3.1 can perform their required safety function(s), including requirements to perform their functions, when specified in the applicable GDCs or PDCs, with onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) and assuming a single failure (when applicable). As described in Section 15.1.3.5 of this report, the baseline AOO (BL-AOO) analysis presented in PSAR Section 15.5.3 uses best-estimate input parameters, and PSAR Section 15.2.1.1, “Baseline Deterministic Safety Analysis,” states its purpose is to model the expected plant response without the postulation of failures. While this type of analysis is helpful to understand the nominal plant response, it does not demonstrate the relation between the design bases of the facility and the GDCs (or PDCs, as applicable) in a consistent manner. For example, PSAR Section 3.1.4.5, “GDC 34 – Residual Heat Removal,” states that the ICS is the only system necessary for residual heat removal capability for ensuring the SAFDLs are met and RCPB design conditions are not exceeded; however, the AOO analysis for generator load reject/turbine trip (PSAR Section 15.5.3.2.1) credits the turbine bypass system for controlling pressure and removing decay heat. PSAR Section 15.5.4.2.1 does analyze a generator load rejection with credit for ICS and failure of the DL2 functions performed by the turbine bypass system and the anticipatory trip system scram functions, but does not apply AOO SAFDL acceptance criteria. Therefore, the PSAR does not demonstrate that SAFDLs are met using only the ICS system for residual heat removal or, alternately, include the turbine bypass system as an SSC needed to meet GDC 34. Similarly, PSAR Section 15.5.4.4.1 describes a loss of all FW postulated accident; however, a loss of all FW (which the staff considers to be an AOO initiating event) is not analyzed as an AOO using the SAFDL acceptance criteria. In both cases, the CN-DSA documented in PSAR Section 15.5.4 analyzes these initiating events; however, because the applicant categorizes these transients as accident event sequences, the results are compared against postulated accident acceptance criteria instead of applicable AOO figures of merit required in the GDCs.

In a letter dated March 2, 2026 (TVA 2026-TN13052), the applicant states that all of the pressure increase DBAs (presented in Section 15.5.4) have similar heat loads when ICS is initiated and show that ICS is capable of removing residual heat and keeping the peak pressure to values below the design limit, even when only one of three trains is credited. The response further concludes that PSAR Section 3.1.4.5 for GDC 34 is acceptable because it is consistent with other approved applications and NRC guidance, and ICS performance is demonstrated under the most limiting DBAs. The staff does not agree that the analysis presented in Chapter 15.5.3 is consistent with the GDC 34 compliance statements in PSAR Section 3.1.4.5, other approved applications, and NRC guidance (SECY-94-084 and its associated staff requirements memorandum [SRM]). Specifically, the response conflates the basis for staff approval of the Economic Simplified Boiling Water Reactor (ESBWR) design control document (DCD), and may not appropriately represent the NRC regulatory treatment of non-safety systems (RTNSS) policies related to active systems. These systems are identified using a different process than what is used to determine the scope of SSCs needed to demonstrate the GDCs, including SAFDL acceptance criteria, are met. However, the staff does agree that the pressure increase events documented in PSAR Section 15.5.4 can be used to infer adequate residual heat removal by the ICS consistent with the applicant's proposed approach to meeting GDC 34. While PSAR Section 15.5.4 does not analyze or document satisfaction of the AOO acceptance criteria, this justification provides sufficient basis that the design can meet GDC 34 in accordance with 10 CFR 50.34(a) and supports issuance of a CP in accordance with 10 CFR 50.35. Consistent with 10 CFR 50.35, the AOO analysis demonstrating the ICS capability to perform its stated GDC 34 safety function can reasonably be left for later consideration to be reviewed with the FSAR at the OL licensing phase.

Related to the absence of an AOO analysis for the loss of FW event, in letter dated March 2, 2026 (TVA 2026-TN13052), the applicant includes a footnote that states the loss of FW event documented in PSAR Section 15.5.4.4.1 does not challenge the SAFDLs. While the staff does not have a reason to believe the AOO acceptance criteria would be exceeded, the applicant will need to demonstrate in the FSAR that the SAFDLs are not challenged using AOO acceptance criteria and associated figures of merit. Consistent with 10 CFR 50.35, the AOO analysis for a loss of FW or justification of why an existing documented AOO within the same event type is bounding, can reasonably be left for later consideration to be reviewed in the FSAR. See Section 15.5.3.3.3 of this report for the staff's evaluation of the BL-AOO loss of a single FW pump.

Single Failure Requirements

The staff also evaluated whether the Chapter 15 safety analysis considered the single failure criterion capability as required by the GDCs. Within this context, a single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions and is separate from the failure that caused the initiating event. PSAR Section 15.2.1.2 states that the single failure criterion is applied to SC1 SSCs. The staff understands that the analysis for postulated accidents only credits SC1 SSCs; however, the PSAR does not specify the single failure criterion was considered in the safety analysis of AOOs. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary) the applicant states DL2/Safety Category 3 functions are not designed to withstand single failures in the same way as the safety-related DL3/Safety Category 1 functions. Further, the applicant stated that the Safety Strategy requires DL2/Safety Category 3 functions to be able to achieve a minimum reliability of 1×10^{-2} failures per demand. The applicant noted that two layers of defense for each AOO PIE (one layer, DL2, having a high reliability requirement and the second layer, DL3, being fully single failure compliant) provide a more reliable AOO response. As described earlier in this report, the staff has not yet approved or endorsed the BWRX-300 Safety Strategy, and its use does not equate to regulatory compliance. Additionally, as noted in Section 15.1.3.1, "Use of Reliability Targets," of this report, specific topics related to the use of DL reliability targets will need to be addressed in the FSAR. Several GDCs contained in 10 CFR Part 50, Appendix A, also require SSCs performing certain AOO and accident prevention and mitigation functions to be designed to withstand single failures. In addition, the GDCs include provisions that the SAFDL acceptance criteria can be met following an AOO assuming a single failure. For example, GDC 20 states that the protection system be designed to assure that SAFDLs are not exceeded as a result of an AOO, and GDC 21 states that the protection system be designed with redundancy and independence to assure that no single failure results in loss of protective function. However, in the April 1, 2026, letter (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary), the applicant states that assuming single active failures in AOO analyses is not required to meet GDC 21 because non-safety-related SSCs supporting DL2 functions needed to meet GDC 21 have "acceptable reliability of operation." The staff disagrees with this interpretation of GDC 21 because the protection system design provisions described in

GDC 21, including single failure, also apply to the protection system functions needed to meet SAFDL acceptance criteria following an AOO as described in GDC 20 and other criteria. Single failure attributes of the protection systems for compliance with GDC 21 is evaluated in Chapter 7 of this report. Because complete design information is not required under the NRC's regulations for issuance of a CP, the staff will review the final design to confirm the safety analysis appropriately considers the single failure requirements of the GDCs (and PDCs as applicable) in the FSAR.

Related to postulated accident scenarios, the staff reviewed whether all pertinent single failures have been considered. PSAR Section 15.5.4 event sequence summary and results sections only document single failures of the ICS. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary), the applicant stated this is due to the relative simplicity of the BWRX-300 design and the benefits of the BWRX-300 Safety Strategy. While the applicant will need to describe the methodology used to select limiting single failures in the FSAR, the staff concluded that the information provided in the PSAR and supporting documentation is consistent with the requirements in 10 CFR 50.34(a) and sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design as required by 10 CFR 50.35. Therefore, this preliminary analysis supports issuance of a CP in accordance with 10 CFR 50.35 and further technical information required to demonstrate that the limiting single failure is considered can be reasonably left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

Electrical Power Requirements

The staff evaluated whether the Chapter 15 safety analysis considered electrical power availability and capability as required by the GDCs/PDCs, and whether it was performed in a manner consistent with the stated design basis in PSAR Section 3.1.2.8, "PDC 17 – Electric Power Systems." Related to AOOs, PSAR Section 3.1.2.8 states that Safety Category 3 functions ensuring fission product barrier design limits are not exceeded as a result of AOOs are provided by the SC3 Preferred Power 250VDC Subsystem in the event of a loss of offsite AC power. Based on the staff's review of the AOO analysis presented in PSAR Section 15.5.3, only the loss-of-preferred power (LOPP) as the assumed initiating event considers the loss of onsite or offsite AC power systems. The safety analysis that demonstrates that the SC3 Preferred Power 250VDC Subsystem alone is adequate for ensuring the fission product barrier design limits (e.g., the SAFDLs for the fuel fission product barrier) are not exceeded can reasonably be left for later consideration in accordance with 10 CFR 50.35 and will be reviewed in the FSAR.

Related to postulated accidents, PSAR Section 3.1.2.8 identifies the SC1 Emergency Power System (i.e., onsite DC power) as the only electrical system relied on to mitigate design-basis accidents in the event of a loss of offsite AC power. Additionally, PSAR Section 8.1, "Electric Power – Introduction," states, [t]he passive design of the plant is not dependent upon offsite or standby diesel generator AC power to mitigate a Design Basis Accident (DBA)." However, based on its review of PSAR Chapter 15, the staff could not confirm this statement because it is unclear how AC power is treated in the analysis of DBA PIEs and if the limiting AC power availability conditions were appropriately captured. Power availability assumptions can have a significant impact on the outcome of each design-basis scenario, but the staff observes that only LOCA DBA scenarios considered both AC power availability and unavailability. In letter dated March 2, 2026 (TVA 2026-TN13052), the applicant stated that assuming a LOPP concurrent with non-LOCA DBA events potentially changes the event sequence category and is not necessary to meet GDC. The staff considered the applicant's response and, as previously discussed in Section 15.1.3.1 of this report, in accordance with 10 CFR Part 50, the event category is based on the initiating event frequency, and considering the reliability provisions specified in the GDCs (such as single failure or onsite/offsite electrical power requirements) does not influence or result in changes to the event category (i.e., an AOO to a DBA or a DBA to a DEC). Further, the stated design basis associated with the design-specific PDC 17 specifies that the only required electrical power system for mitigating DBAs is the SC1 DC power system. With the exception of the LOCA analysis presented in PSAR Section 15.5.4.5, “Loss of Coolant Accidents Design Basis Accidents,” the safety analysis of postulated accidents presented in PSAR Chapter 15 does not consider the loss of AC power coincident with the initiating event or is performed in a manner that only relies on the SC1 DC power system. However, the staff concluded that the information provided in the PSAR and supporting documentation is sufficient to meet the requirements in 10 CFR 50.34(a) and supports issuance of a CP in accordance with 10 CFR 50.35. Further technical information required to complete the safety analysis and demonstration of the SC1 Emergency Power system to perform its design basis safety functions in PDC 17 can be reasonably left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

For the onsite and offsite power systems, the staff will review the final design’s compliance with the electric power portions of GDC 33, 34, 35, 38, 41, and 44 in the FSAR and confirm that the safety functions prescribed in these GDCs can be accomplished by the SSCs identified in PSAR Section 3.1, because complete design information is not required under the NRC’s regulations for issuance of a CP.

Setpoints and Time Delays

SRP Section 15.0 specifies that the reviewer ascertains that the application lists the expected limiting delay time for each protection or safety system function and describes the acceptable methodology for determining uncertainties (from the combined effects of calibration error, drift, instrumentation error, and other factors). This information is needed to support the establishment of the trip setpoints and allowable values specified in the plant technical specifications. PSAR Table 15.5-5 includes a listing of the assumed setpoints and signal delay times. The “System Design Bases and Associated Safety Functions” described in PSAR Sections 7.3.1.2 and 7.3.3.2 state, “the setpoints for SC1 and SC3 I&C systems (i.e., those systems credited for mitigating design basis AOOs and accidents) are determined with the setpoint methodology defined in IEC 61888, “Nuclear Power Plants - Instrumentation Important to Safety – Determination and Maintenance of Trip Setpoints,” using the final analytical limits from the plant safety analyses and the measurement uncertainties associated with the I&C equipment (e.g., sensors and processing units), respectively. The staff finds that identifying expected setpoints and associated time delays in the PSAR conforms to the SRP guidance and supports issuance of the CP. The staff will evaluate the applicant’s setpoint methodology and final values as part of its review of the FSAR. The staff will evaluate the FSAR to ensure the SC1 I&C systems will be able to automatically actuate, correct the abnormal situation, and complete the protective function before a safety limit is reached or exceeded consistent with 10 CFR 50.2 and 10 CFR 50.36(c)(1)(ii)(A).

Summary

Based on the above, the staff finds that the applicant has described the proposed design of the facility consistent with the requirements in 10 CFR 50.34(a) and identified the major features or components incorporated therein for the protection of the health and safety of the public with sufficient detail to support issuance of a CP in accordance with 10 CFR 50.35. The staff has noted several instances where further technical or design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase as specified in 10 CFR 50.35(a)(2). This includes assessment of AOOs with appropriate design basis assumptions demonstrating the safety systems identified PSAR Section 3.1, “Compliance with U.S. Nuclear Regulatory Commission General Design Criteria,” can perform their required safety functions and demonstrate that appropriate AOO analysis acceptance criteria can be met.

15.1.3.7 Analytical Methods

Section 15.5.1.3 of this report provides a description of the computer codes used by the applicant to perform the preliminary safety analysis documented in PSAR Chapter 15.

The staff used the guidance provided in SRP Section 15.0 to support the review of the applicant’s analytical methods. The staff assessed the applicant’s use of topical reports (TRs) that describe models or computer codes used in transient and accident analyses and conformance with the associated NRC safety evaluation reports approving those TRs, where applicable. In addition, the staff verified that implementation of NRC-approved models or codes are within the applicable ranges and conditions and that the applicant has demonstrated compliance with each of the NRC specified conditions and limitations associated with approved TRs.

PSAR Tables 1.9-15 and 1.9-20 identify that the applicant intends to conform to SRP Section 15.0.2, “Review of Transient and Accident Analysis Methods,” and RG 1.203, “Transient and Accident Analysis Methods,” respectively. SRP Section 15.0.2 and RG 1.203 provide acceptable methods for the analysis of accident and transient behavior and for determining whether each of the evaluation models meet the requirements of 10 CFR 50.34 and 10 CFR 50.46. The staff reviewed applicant’s approach for performing DSA and the transient and accident analysis results presented in Chapter 15 of the PSAR. The staff found that the applicant generally used established computer codes and described their preliminary analysis methods in the PSAR. The staff notes that the analytical approach used in the PSAR Section 15.5 analysis did not address all elements described in RG 1.203 and SRP Section 15.0.2. Gaps included the power availability assumptions and the application of single failure to AOO analysis (discussed in Section 15.1.3.6 of this report), application of a containment response analytical methodology to the assessment of fuel cooling performance (discussed in Section 15.5.4.3.6 of this report), and applicability and implementation of the TRACG computer code to the BWRX-300 design (discussed in Section 15.11.3 of this report).

At the CP stage, the staff’s review focuses on determining whether the application establishes the proposed design of the facility, including the principal architectural and engineering criteria for the design, and has identified the major features necessary for the protection of the health and safety of the public. Consistent with 10 CFR 50.34(a)(4), the analyses presented in PSAR Chapter 15 are considered preliminary and are intended to demonstrate that the plant design can be evaluated against the appropriate transient and accident acceptance criteria. While the staff has determined that the preliminary analyses are consistent with the requirements in 10 CFR 50.34(a) and are sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design as required by 10 CFR 50.35, the staff notes that the analytical methodologies used to perform these analyses have not been approved by the NRC for application to this design for these events and figures of merit. As described in 10 CFR 50.46 for LOCA evaluation models and other guidance for non-LOCA evaluation models, safety analyses used to demonstrate compliance with the applicable acceptance criteria should be performed using NRC-approved evaluation models and analytical methodologies, or the applicant must provide sufficient information for the NRC staff to review and approve the methodology for application to a given design. As discussed in Section 15.5.4.3.6 of this report, the staff has initiated an exemption to the requirement to perform an analysis and evaluation of ECCS cooling performance in accordance with an acceptable evaluation model. For the purposes of the CP review, the staff performed a preliminary assessment of the neutronic and thermal-hydraulic computer codes and analysis approaches related to the fuel and RCS fission product barrier figures of merit for the preliminary safety analysis. For the staff's specific assessment of how these models support issuance of a CP, see Section 15.11 of this report for the staff's review of the applicant's preliminary analytical methodology and use of TRACG for the performance of Chapter 15 non-LOCA events and see Section 15.5.4.3.6 of this report for the staff's review of the applicant's preliminary LOCA evaluation model.

For the OLA, the applicant must provide the final Chapter 15 safety analyses demonstrating that the plant meets the applicable acceptance criteria for AOOs and DBAs, including LOCAs. These analyses must be performed using evaluation models and analytical methodologies that have either been approved by the NRC for application to the BWRX-300, or sufficient information must be provided in the OLA to support the staff's review of the evaluation models. The information in the evaluation model should specify figures of merit used for the safety analysis, justification that analysis methods are applied within the applicable ranges and conditions, and demonstration that conditions and limitations imposed by the NRC staff in approved LTRs are met, as applicable. Also, as discussed throughout this report, the analytical methods need to account for the permitted fluctuations in the expected operating band and associated uncertainties in input parameters and initial conditions for all DBEs. See Section 15.11 of this report for a detailed explanation of what constitutes an acceptable evaluation model for the purposes of performing design-basis transient and accident analysis.

Related to the assessment of the containment fission product barrier, the applicant utilized an NRC-approved methodology in its PSAR. See Section 15.5.5.4.3.5 of this report for the staff's evaluation of containment performance.

15.1.4 Conclusion

The NRC staff has reviewed the applicant's approach to preliminary transient and accident analyses provided in PSAR Sections 15.1 through 15.4 against the applicable requirements of 10 CFR Part 50, including 10 CFR 50.34(a), 10 CFR 50.46, 10 CFR 50.35, and Appendix A to 10 CFR Part 50, and the guidance in SRP Chapter 15. Consistent with the scope of a CP review under 10 CFR 50.35, the staff's evaluation at this stage does not reach final findings on the adequacy of the applicant's analytical methods, event categorization scheme, acceptance criteria, or system design bases. Instead, the staff's determination is limited to assessing whether the application contains sufficient preliminary design information and an analytical framework that will allow the staff to complete the safety review during the OL stage.

Based on the staff's evaluations of individual events and event types within the preliminary deterministic safety analysis in Section 15.5 of this report, the staff determined that the safety analyses presented in Chapter 15 of the PSAR are consistent with the requirements in 10 CFR 50.34(a) and adequately demonstrate the principal architectural and engineering criteria of the design as required by 10 CFR 50.35. Therefore, the preliminary analyses are sufficient to support issuance of a CP in accordance with 10 CFR 50.35 and, as noted in several areas in the discussions above, further technical and design information needed to complete the safety analysis can be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.2 Identification, Categorization, and Grouping of Postulated Initiating Events and Accident Scenarios

15.2.1 Introduction

Section 15.2 of the PSAR describes the process used to identify, categorize, and group PIEs and accident scenarios for the BWRX-300 at the CRN Site. The purpose of this section is to ensure that the full spectrum of potential events that may affect plant safety is systematically evaluated in support of the facility's safety analyses.

15.2.2 Regulatory Evaluation

The regulatory basis described in Section 15.1.2 of this report also applies to PSAR Section 15.2.

15.2.3 Technical Evaluation

The technical evaluation of PSAR Section 15.2 is provided in Section 15.1.3 of this report.

15.2.4 Conclusion

The staff conclusion for the evaluation of PSAR Section 15.2 is provided in Section 15.1.4 of this report.

15.3 Safety Objectives and Acceptance Criteria

15.3.1 Introduction

Section 15.3.1 of the PSAR outlines the safety objectives and acceptance criteria used to evaluate the deterministic safety analyses of the CRN-1 BWRX-300 reactor design at the Clinch River Nuclear Site. Derived qualitative and quantitative acceptance criteria used to analyze AOOs and DBAs are provided in PSAR Table 15.3-1 and Table 15.3-2, respectively.

PSAR Section 15.3.2 presents the acceptance criteria for probabilistic risk analysis for CDF and LRFs, consistent with the Commission's Safety Goals Policy Statement, "Safety Goals for the Operation of Nuclear Power Plants" (51 FR 28044-TN12078; August 4, 1986) (Safety Goals Policy Statement). PSAR Table 15.3-3 presents the probabilistic safety goals that supplement the deterministic criteria.

15.3.2 Regulatory Evaluation

The regulatory basis described in Section 15.1.2 of this report also applies to PSAR Section 15.3.1 for the deterministic acceptance criteria. Probabilistic safety assessment acceptance criteria are presented in PSAR Section 15.3.2 and PSAR Table 15.3-3, "Probabilistic Safety Goals." The applicant stated that these criteria are based on the Commission's Safety Goals Policy Statement. The staff notes that the Commission approved subsidiary numerical objectives for CDF and LRF in SRM-SECY-90-016 to support implementation of the health objectives in the Commission's Safety Goals Policy Statement.

These numerical objectives address CDF and LRF for new LWRs. In addition, in SRM-SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs,” dated July 21, 1993 (NRC 1993-TN12041), the Commission approved containment performance goals. Collectively, these numerical goals limit individual and societal risk. The use of PRA to inform regulatory decisions is supported by the Commission’s policy on the “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities” (60 FR 42622-TN6278; August 16, 1995). The staff’s evaluation ensures that the proposed design is consistent with these safety goals and provides adequate protection of public health and safety. SRP Chapter 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” provides guidance for reviewing the use of PRA for new reactor licensing applications.

15.3.3 Technical Evaluation

The technical evaluation of the deterministic acceptance criteria contained in PSAR Section 15.3 is provided in Section 15.1.3.5 of this report.

The staff notes that in PSAR Table 1.9-19, “Conformance with NUREG-0800 (Chapter 19 Severe Accidents),” the applicant notes that conformance with SRP Section 19.0 will be deferred to the FSAR. 10 CFR 50.34(a) (TN249) does not require an applicant for a CP to develop or submit a PRA or its acceptance criteria. Therefore, the applicant’s probabilistic safety assessment acceptance criteria will be reviewed at the OL licensing phase.

15.3.4 Conclusion

The staff’s review of the deterministic acceptance criteria contained in PSAR Section 15.3 is provided in Section 15.1.3.5 of this report.

The staff acknowledges that TVA included the PRA acceptance criteria in its PSAR; however, the staff has not evaluated this information as part of its safety review to support the issuance of the CP as 10 CFR 50.34(a) does not require an applicant for a CP to develop or submit a PRA or its acceptance criteria. Accordingly, the staff does not take a position on the information presented in this section related to the PRA acceptance criteria.

15.4 Human Actions

15.4.1 Introduction

Section 15.4 of the PSAR describes the role of human actions in the safety analyses for the BWRX-300 design proposed for the CRN Site. This section provides an overview of how the design minimizes the potential for human error, the treatment of operator actions in deterministic and probabilistic safety analyses, and the methodology used to identify and evaluate human actions that may affect plant safety.

15.4.2 Regulatory Evaluation

The staff based its review on the following regulations and guidance:

- 10 CFR 50.34(a)(4) (TN249)
- SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs” (NRC 1994-TN12883)

15.4.3 Technical Evaluation

The applicant stated in PSAR Section 15.4.1, “Human Actions in Deterministic Safety Analysis,” that for PIEs and event sequences in the AOO or DBA event categories, the Fundamental Safety Functions can be performed and maintained for 72 hours without operator action and that this is substantiated in the CN-DSA. Section 15.5 of this report provides the staff’s evaluation of individual events and event types analyzed within the preliminary DSA. In addition, the HFEA identifies failures that involve an erroneous decision or action taken by a human that can potentially initiate a PIE. In PSAR Section 15.1.2.2, the applicant clarified that the inputs needed to complete the HFEA are incomplete at the PSAR stage and will be complete by the FSAR. Human actions resulting from PSA event evaluations are discussed in PSAR Section 15.6.3, which was not reviewed as part of the CP evaluation as the PSA is not required by 10 CFR 50.34(a) for issuance of a CP (see Section 15.6 of this report). Once the final design’s safety analysis is complete, the staff will evaluate the overall ability of the BWRX-300 design to mitigate DBEs without operator action during the FSAR review at the OL stage.

15.4.4 Conclusion

The staff has not evaluated information related to human actions as part of its safety review to support the issuance of the CP, as this evaluation is not required for issuance of a CP. Human actions credited in the safety analysis will be reviewed at the OL stage.

15.5 Deterministic Safety Analyses

15.5.1 General Description of the Approach

15.5.1.1 Introduction

Section 15.5.1 of the PSAR presents the safety margins, analytical codes, and standards employed in the DSAs that support the BWRX-300 design proposed for the CRN Site. DSA is performed to evaluate the performance of the engineered barriers (e.g., fuel cladding, reactor pressure boundary, containment boundary) during different events and calculate the dose consequences of breaching those barriers in the events. This section provides a summary of the methodologies, analytical tools, and assumptions used to evaluate the plant’s response to a set of PIEs, including AOOs, DBAs, and DECs, as identified and grouped in Sections 15.1 and 15.2 of the PSAR. The AOOs, DBAs, and DECs rely on DL2, DL3, and DL4 functions to mitigate the events.

15.5.1.2 Regulatory Evaluation

The staff based its review on the following regulations and guidance:

- 10 CFR 50.34(a), “Contents of application, technical information – Preliminary safety analysis report” (TN249)
- 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors”
- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants”
- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition”, Sections 15.0 and 15.0.2 (NRC 2021-TN8013)

15.5.1.3 Technical Evaluation

PSAR Section 15.5.1 provides a general description of the approach for the DSA. Events are divided into non-LOCA and LOCA events. The non-LOCA events can be in any category (AOO, DBA, DEC) and evaluate the integrity of the fuel cladding and reactor pressure boundary barriers. The LOCA events evaluate the integrity of the fuel cladding and containment pressure boundary barriers assuming a breach of the reactor pressure boundary barrier. Both the non-LOCA and LOCA events are categorized as large, medium, or low safety margin events, and the margin determines how initial conditions and the evaluation of uncertainties are treated in the calculations. This is discussed and reviewed in Section 15.11 of this report. An evaluation of the actual events follows in later subsections of Section 15.5 of this report.

PSAR Section 15.5.1.2 provides an overview of some of the important specific analysis tools that are used in the deterministic analysis. The codes described that are used in the evaluation of the boundary integrity are TRACG, GOTHIC, and PANAC11. The codes described that are used in the dose consequence calculations are ANSI/ANS-18.1-2020 Standard, PAVAN, RADTRAD, ARCON, and ORIGEN2. This section of the SER provides a summary of the primary codes used in the DSA. For those codes or methods where additional evaluation is performed, the staff provides a reference to the section of this report where that evaluation resides. Otherwise, the staff is not required to make any specific findings in this section to support issuance of a CP in accordance with 50.34(a).

TRACG

TRACG is a multi-physics thermal-hydraulic systems safety analysis code that models two-phase compressible flow coupled to heat structures. The fuel rod heat structures have thermo-mechanical models and can use either point or multi-dimensional reactor kinetics. The gas phase can be a mixture of steam and non-condensable gas, and the liquid phase can contain dissolved boron. Flow regime dependent constitutive equations (drag and heat transfer correlations) are required for the coupling between the fluid phases and between the fluid phases and the heat structures. Equations of state for the liquid and gas are required to close the system of equations. TRACG has a wide validation range for steady state and transient conditions including LOCA conditions against separate effects tests, component tests, integral effects tests, and operating plant data. NEDC-34043P, “BWRX-300 TRACG Application,” provides additional details about how TRACG is applied to BWRX-300 transient and LOCA analysis. The GVH NEDC-34043P engineering report has not been previously reviewed and approved by the staff. The staff’s evaluation of NEDC-34043 and its applicability to the BWRX-300 PSAR is discussed in Section 15.11 of this report.

GOTHIC

GOTHIC is a containment analysis code that has both lumped parameter and three-dimensional (3D) finite volume modeling capabilities. It is used in the 3D mode for modeling the performance of the BWRX-300 containment in response to a LOCA. GOTHIC is a commercially available computer code developed for the Electric Power Research Institute (EPRI) that is used by many organizations for containment safety analysis and has a wide validation range. Most previous containment analysis methods used by GVH were developed for BWR pressure suppression containments, and those methods have not been shown to be applicable to the new containment design used by the BWRX-300, which is similar to PWR dry containments. GOTHIC has been widely used to analyze PWR dry containments. The NRC-approved methodology for applying GOTHIC to BWRX-300 LOCA containment analysis is described in

NEDC-33922P-A Revision 3, “BWRX-300 Containment Evaluation Method.” TRACG is used to provide the mass and energy source terms for the containment analysis in the methodology. The approval was subject to four limitations and conditions (L&Cs) discussed in PSAR Section 15.5.4.5, “Loss of Coolant Accidents Design Basis Accidents.” The staff analysis of whether and how those conditions are met for the PSAR design is discussed in Section 15.5.4.3.5 of this report.

PANAC11

PANAC11 is a 3D nodal core simulator that uses a coupled neutronics and thermal-hydraulics model to model the reactor core throughout the fuel cycle. It provides the core conditions at different points in the fuel cycle (beginning, middle, and end of cycle [resp. BOC, MOC, and EOC]) to TRACG to use in transient calculations in Chapter 15. The 3D nodal neutronics model in TRACG is consistent with the model in PANAC11.

ANSI/ANS-18.1-2020 Standard

The applicant determined radiation concentrations in the reactor coolant and steam during normal operations using the American National Standards Institute (ANSI)/American Nuclear Society (ANS)-18.1, “Radioactive Source Term for Normal Operation of Light Water Reactors.” The staff evaluates the use of this standard in Section 11.1.3 of this report.

PAVAN

The applicant used the computer code PAVAN (NUREG/CR-2858, “PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations”) to estimate atmospheric dispersion χ/Q values at the exclusion area boundary (EAB) and at the outer boundary of the low population zone (LPZ) for potential accidental releases of radioactive material. The PAVAN model implements the methodology outlined in RG 1.145 (NRC 1983-TN279), Revision 1, as described in Subsection 2.3.4.2 of the Site Safety Analysis Report for the Clinch River Nuclear Site Early Site Permit Application, Revision 2 (TVA 2019-TN5854). The PAVAN code estimates χ/Q values for various time-average periods ranging from 2 hours to 30 days. A straight-line trajectory is assumed between the point of release and all distances for which χ/Q values are calculated. For each of the 16 downwind direction sectors (e.g., N, NNE, NE, ENE), PAVAN calculates χ/Q values for each combination of wind speed and atmospheric stability at the appropriate downwind distance (e.g., the EAB and the outer boundary of the LPZ).

RADTRAD

The applicant determined the dose consequences of postulated DBAs using the RADTRAD Version 3.10 computer code as described in NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation.” The RADTRAD code was developed by the Accident Analysis and Consequence Assessment Department at Sandia National Laboratories for the NRC, Office of Nuclear Reactor Regulation, Division of Reactor Program Management. If deemed necessary to make a reasonable assurance finding, the staff can perform independent confirmatory radiological calculations using the RADTRAD computer code, run within the Symbolic Nuclear Analysis Package (SNAP) suite of integrated applications for engineering analysis, developed for the NRC. Information on the SNAP/RADTRAD code is available from the NRC’s Radiation Protection Computer Code Analysis and Maintenance Program at https://ramp.nrc-gateway.gov/content/snapradtrad-overview.

ARCON

The ARCON Code, ARCON, NUREG/CR-6331, “Atmospheric Relative Concentrations in Building Wakes,” is used to model radionuclide transport to the onsite main control room envelope (MCRE) and offsite EAB and LPZ locations. The ARCON code was developed by Pacific Northwest National Laboratory for the NRC, Office of Nuclear Reactor Regulation, Division of Reactor Program Management.

ORIGEN2

The applicant used the ORIGEN2, Version 2.1, “Isotope Generation and Depletion Code – Matrix Exponential Method,” computer code system for calculating the buildup, decay, and processing of radioactive materials. ORIGEN2 is typically used for estimating core inventories of nuclides for LWRs at various stages of power operation. The ORIGEN code is widely used in the nuclear industry to calculate fission product production and depletion and is endorsed in regulatory guidance for the calculation of core fission product inventory.

15.5.1.4 Conclusion

This section provides a description of the applicant’s approach for performing the DSA and provides an overview of the codes and methods used in the DSA. The staff is not required to make a regulatory finding in accordance with 10 CFR 50.34(a) for issuance of a CP on this general description of the analysis approach or the overview of codes and standards.

15.5.2 BWR Stability Analysis

15.5.2.1 Introduction

This section of the PSAR describes the applicant’s evaluation of BWR stability for the BWRX-300 design to be constructed at the CRN Site. The stability analysis addresses the ability of the proposed BWRX-300 design to preclude undamped power oscillations and other thermal-hydraulic instabilities under AOOs and postulated accident conditions. The applicant’s submittal provides analytical methods, criteria, and results that demonstrate compliance with applicable regulatory requirements and guidance for BWR stability, including conformance with the guidance in SRP Section 15.9, “BWR Stability,” and the methodologies described in NEDO-31960-A, “Licensing Topical Report, BWR Owners’ Group Long-Term Stability Solutions Licensing Methodology” (GE 1995-TN13154), and its supplement (GE 1995-TN13155).

15.5.2.2 Regulatory Evaluation

The staff based its review on the following regulations and guidance:

- GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs.
- GDC 12, “Suppression of Reactor Power Oscillations,” as it relates to the RCS and protection systems being designed to assure that power oscillations which can result in conditions that exceed SAFDLs are not possible or can be detected and suppressed.
- NUREG-0800, SRP Section 15.9, “BWR Stability,” which the applicant states they conform with as noted in PSAR Table 1.9-15, “Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)” (NRC 2021-TN8013).

15.5.2.3 Technical Evaluation

The applicant briefly summarized the results of BWRX-300 stability analysis performed for nominal rated conditions with an equilibrium cycle core using GNF-2 fuel. The staff reviewed these results, given in Chapter 15.5.2, "BWR Stability Analysis," Table 15.5-2, and Figures 15.5-1 through 15.5-3 of the PSAR. The applicant leveraged the methodology and results documented in NEDC-34270P, Revision 1, "BWRX-300 Stability Analysis" (GE Verona 2026-TN13153 [non-proprietary], ML26089A384 [proprietary]), which is referenced and described in Appendix 4A, "Thermal-Hydraulic Stability," of the PSAR. Based on the stability analysis results, the applicant concluded that the stability considerations during normal operation and AOOs meet the intent of SRP Section 15.9. In PSAR Table 1.9-15, the applicant listed the stability analysis conformances with SRP Section 15.9 using the following eleven criteria:

1. Design to be free of undamped oscillations and thermal hydraulic instabilities
2. Detection and suppression of oscillations
3. Methodologies provided in NEDO-31960 and its Supplement 1
4. Stability criteria (maximum calculated decay ratio of 0.8 with acceptable uncertainties)
5. Operating exclusion area
6. Reactor scram before SAFDL violation
7. Stability modeling
8. Backup options; Technical Specification requirements
9. New stability solutions
10. Stability related instrumentation functionality
11. Instability modes other than density wave instability

As discussed in Chapter 4, Section 4A of this report, the staff evaluated the application's conformance with the above 11 criteria. The first criterion requires the reactor to be free of undamped oscillations and thermal-hydraulic instabilities. GVH indicated in NEDC-34270P that thermal-hydraulic instabilities are inevitable during start-up. As discussed in Chapter 4, Section 4A of this report, the staff finds the stability method sufficient for the purposes of CP issuance, and that the analytical results and technical justification of compliance with GDCs 10 and 12 during startup conditions can reasonably be left for later consideration and will be supplied in the FSAR.

The second criterion is the detection and suppression of oscillations. The BWRX-300 preliminary design does not incorporate a special stability detection and trip system. Since the staff agrees that the design is not susceptible to regional-mode oscillations, and the criteria specified in Appendix 4A of the PSAR are that any operational perturbation, maneuver, or transient either results in an immediate scram or oscillations that decay quickly and protect the SAFDLs, it is acceptable that the preliminary design does not include detection and suppression measures.

The methodologies provided in NEDO-31960 and its Supplement 1 listed as criterion 3 were developed to resolve operating BWR stability issues. Their long-term solution and detection/suppression measures are not adopted by the BWRX-300 design. Therefore, this criterion is not applicable.

The fourth criterion is satisfied through the specification of stability criteria of a decay ratio less than or equal to 0.8, which is satisfied on a preliminary basis for the core-wide and regional Type 2 instabilities under nominal rated operating conditions. In its evaluation of the results presented in PSAR Section 15.5.2, the staff notes that these preliminary results show the decay ratio for the core-wide oscillation mode is less than the acceptance criteria of 0.8. This demonstrates that the BWRX-300 design appears to have sufficient stability margin against core-wide density wave oscillations under nominal rated operating conditions. This margin must be demonstrated and confirmed with each new cycle design following NEDC-34270, Revision 1. The stability analysis methodology in NEDC-34270, Revision 1, includes the necessary provisions to address uncertainties associated with cycle specific conditions. The FSAR submitted with the OLA must include a calculation using the initial cycle design to confirm the core-wide decay ratio remains below the acceptance criteria.

In NEDC-34270, Revision 1, the calculated regional instability decay ratio is much smaller than 0.8. This further supports the applicant's claim that the core-wide oscillation is dominant. Therefore, the staff agrees that calculations to confirm regional instability is not present are not necessary for the Clinch River BWRX-300 reactor.

Relative to Criterion 5, PSAR Section 4.4 includes Table 4.4-2, specifying the expected BOC core flow and nominal FW temperature versus power. This relationship is central to the underlying assumptions in the stability analysis. The staff's evaluation of how this criterion will be met is discussed in Chapter 4, Section 4A of this report.

The required scram before SAFDL violation specified in Criterion 6 has been preliminarily demonstrated for the most limiting loss of FW transient starting from nominal rated operating conditions. For all other AOOs initiated from off-rated conditions, this conformance has not been demonstrated. Therefore, the FSAR submitted with the OLA must include an evaluation of stability margins for AOOs initiating from off-rated conditions, as well as from the full power condition, to support final findings on GDCs 10 and 12. The evaluation must also justify whether other AOOs besides loss of feedwater heating (LFWH) may be limiting at off-rated conditions.

The stability modeling and the relevant analysis methodology specified in Criterion 7 are documented in NEDC-34270P, Revision 1. The licensing topical report (LTR) proposed an implicit integration scheme using the TRACG code to determine the core-wide decay ratio. This methodology is described in Chapter 4, Section 4A of the PSAR, and evaluated by the staff in Chapter 4, Section 4A of this report.

Criterion 8 is not applicable, as a detection/suppression strategy is not employed with the proposed design. Nonetheless, Chapter 4, Section 4A of this report discusses Technical Specification requirements to ensure the initial conditions relative to the stability analyses are protected.

Since neither new stability long-term solutions nor stability related instrumentation are adopted by the BWRX-300 design, conformance with Criteria 9 and 10 is not needed. Relative to Criterion 11, the staff's evaluation of how Type 1 instability for start-up conditions will be met is discussed in Chapter 4, Section 4A of this report.

15.5.2.4 Conclusion

The preliminary stability analysis results summarized in PSAR Section 15.5.2 indicate a core-wide oscillation decay ratio less than the acceptance criteria of 0.8 under nominal rated operating conditions for an equilibrium cycle design. The stability results were generated by the TRACG code consistent with the methodology in NEDC-34270P, Revision 1.

The staff recognizes that PSAR Appendix 4A and Section 15.5.2 are not complete for staff to make GDC 10 and GDC 12 safety findings. The staff finds that the applicant has identified relevant requirements and provided a sufficient description of preliminary facility design and criteria to authorize construction in accordance with 10 CFR 50.34(a) (TN249). The applicant has identified appropriate regulatory requirements and guidance and a methodology to perform final analyses at the FSAR stage. The staff finds this level of preliminary information sufficient to support issuance of the CP pursuant to the regulations of 10 CFR 50.35.

As described in the staff's evaluation above, and in addition to the evaluations discussed in Chapter 4, Section 4A of this report, the staff finds that further technical information supporting the acceptance of Criteria 1, 5, 8, and 11 can reasonably be left for later consideration, to be reviewed in the FSAR at the OL licensing phase, consistent with 10 CFR 50.35.

15.5.3 Analysis of Anticipated Operational Occurrences

15.5.3.1 Introduction

This section presents the deterministic safety analysis of AOOs for the BWRX-300 design proposed at the Clinch River Nuclear Site, as described in the PSAR. AOOs are defined as those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.

The AOO analyses employ the BWRX-300 Safety Strategy, which is stated to utilize a defense-in-depth approach and incorporates lessons-learned from operating experience, deterministic analysis, and risk-informed methods. The approach for identifying and analyzing AOOs is described in Sections 15.1 through 15.3 of the PSAR and includes the systematic identification of PIEs, event categorization, and grouping based on their frequency and effect on plant parameters.

This section summarizes the scope, methodology, and key results of the deterministic analyses of AOOs, with references to the detailed event descriptions and acceptance criteria provided in Sections 15.2 and 15.3 of the PSAR. The analyses also provide the technical basis for the development of Technical Specifications, as discussed in Chapter 16 of the PSAR.

15.5.3.2 Regulatory Evaluation

The staff based its review on the following regulations and guidance:

- 10 CFR 50.34(a)(4) (TN249)
- 10 CFR 50.35
- 10 CFR 50.46
- GDCs and PDCs identified in Section 15.1.2 of this report, as applicable.

15.5.3.3 Technical Evaluation

In reviewing PSAR Section 15.5.3, the staff has concluded that the AOO analyses presented in the PSAR, and described below, are sufficient to evaluate the design and performance of SCCs and assess the risk to public health and safety resulting from operation of the facility in accordance with 10 CFR 50.34(a). While performing their review, the staff observed that AOO DBEs were analyzed using a best-estimate baseline analysis crediting certain non-safety related but important to safety SSCs and non-safety SSCs rather than using only safety related SSCs for mitigation. As discussed in Section 15.1.3 of this report, the AOO analyses presented in the PSAR did not systematically evaluate initial conditions, power availability, single failure requirements, and safe shutdown fuel performance criteria. However, 10 CFR 50.35 permits issuance of a CP when, among other things, sufficient preliminary information is provided in the CPA and the NRC staff determines that further information required to complete the safety analysis, which can reasonably be left for later consideration, can be supplied in the FSAR. The staff's evaluation and conclusions with respect to the AOO analyses presented in the PSAR are further discussed in the subsections below.

15.5.3.3.1 Decrease in Core Coolant Temperature (AOO) Event Category

In PSAR Section 15.5.3.1, "Decrease in Core Coolant Temperature AOO," the applicant identified LFWH at full power nominal conditions as the bounding decrease in core coolant temperature AOO. In particular, the applicant postulates that the event is initiated by a single failure of either the closure of one extraction steam valve or the inadvertent bypass of a FW heater, and the failure results in the instantaneous decrease in FW temperature. In PSAR Section 15.2.4, the applicant stated that (1) full power conditions are expected to result in the most significant challenge to the fission product barriers and (2) the identification of the most limiting scenarios under all conditions (e.g., low power) cannot confidently be performed until the design is complete and DSA of the final design is performed, which cannot practicably be achieved until the FSAR phase. The staff will confirm that the limiting scenario was identified once final design information is available in the FSAR.

The applicant considered the ICS, control rod drive (CRD) system, reactor water cleanup system (CUW), and shutdown cooling system (SDC) as other systems that could potentially reduce the coolant temperature. Steam pressure regulator malfunctions or failures are captured under the inventory reduction event category. Inadvertent ICS initiation PIEs are evaluated in the increase in reactor coolant inventory fault group. The applicant states that the other systems can only reduce the coolant temperature a small fraction relative to FW related PIEs. PSAR Figure 5.1-5, "BWRX-300 RCS Schematic Flow Diagram at 100% Power," documents the nominal flow rates for the FW, CRD, and CUW systems. The table in PSAR Figure 5.1-5 documents that the nominal FW flow is two orders of magnitude greater than the nominal CUW and CRD flows. Therefore, the staff agrees that the CUW and CRD related temperature decrease PIEs are non-limiting compared to the LWFH PIE. Based on the information in Chapter 15 of the PSAR, the staff could not preclude inadvertent actuation of SDC at power as an AOO PIE due, for example, to an operator error. In PSAR Section 15.1.2.2, the applicant clarified that the inputs needed to complete the HFEA are incomplete at the PSAR stage and will be complete by the FSAR. During the regulatory audit the staff inquired about the expected SDC flows and temperatures required to meet the SDC design requirements in PSAR Section 5.4.7.2, "Shutdown Cooling System – System Description". In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary), the applicant provided preliminary nominal SDC flow rates and SDC heat exchanger outlet temperatures to achieve the design requirements documented in PSAR Section 5.4.7.2. The applicant states that, with initial reactor coolant temperature of 160 degrees Celsius (°C) and corresponding SDC heat exchanger outlet temperature of 97°C, plant cooling water inlet temperatures at the maximum of 35°C and two trains running, SDC flow for each train will be at 270 gallons per minute (gpm) for the first 10 hours. The staff notes that the nominal FW and main steam (MS) temperatures at power conditions are 241.8°C and 284.8°C, respectively (compared to the value in the April 1, 2026, letter of 160°C); therefore, the SDC flow and flow temperature would likely be different if actuated at full power. Staff determined that these SDC nominal parameters indicate that a decrease in FW temperature in the order of 10°C is possible if inadvertent actuation of SDC is credible. In addition, in the regulatory audit (NRC 2025-TN13059), the applicant clarified that the SDC return flow to the CFS is downstream of the FW temperature measurement that informs the DL2-27 function, which indicates that DL2-27 would not be protective for an inadvertent actuation of SDC. PSAR Table 15.5-5 indicates that DL2-27 is designed to detect a feedwater temperature decrease. The staff notes that PSAR Figure 5.4-2 "Shutdown Cooling System Simplified Diagram – Decay Heat Removal," PSAR Figure 5.4-3 "Shutdown Cooling System Simplified Diagram – Overboarding," and PSAR Figure 6.3-1 "Isolation Condenser System Simplified Diagram" show that the SDC suction lines are on the IC trains A and B, outboard of the normally open IC condensate line RIVs and downstream of the normally closed, parallel IC actuation valves. PSAR Section 5.4.7 states that the "SDC is capable of operating at high pressures and temperatures up to the reactor operating pressure and temperature, however SDC operation at these elevated conditions is not expected during normal operation;" and later indicates that under normal operation the SDC has inadequate net positive suction head due to voiding in the chimney section of the RPV. During its regulatory audit, staff observed the PSA system notebook documented an operational restriction that specified the SDC shall not be in service in Mode 1. This is consistent with PSAR section 5.4.7.2.3 which states, "SDC is shut down as the plant heats up during startup operations. The cooling function is no longer required, as heat is being intentionally added to the RPV, and the overboarding function transitions from SDC to CUW when the RPV pressure is sufficient to drive the necessary flow through CUW to the overboarding destination. Upon shutdown, the pump is turned off and the system isolation valves closed." Based on the above, the staff believes the system parameters provided in letter dated April 1, 2026 (ML26091A346, non-proprietary, ML26091A348, proprietary) are for conditions other than power operation (Mode 1). Therefore, the staff concludes for the preliminary design, that inadvertent operation of SDC is likely non-limiting among the other at power TD events.

To verify that the limiting LFWH PIE was identified for nominal full power conditions, the staff requested the applicant perform a sensitivity where the decrease in FW temperature is just less than the DL2-27 (selected control rod run in on feedwater temperature decrease) setpoint. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant provided results for this sensitivity performed under the BWRX-300 stability analysis review. The applicant stated that the sensitivity is not potentially limiting with respect to the operating limit minimum critical power ratio (OLMCPR) or fuel centerline melt and cladding strain criteria, because the event overpower is significantly less than what has been shown to be acceptable for GNF2. In addition, the applicant stated that the FSAR will contain both the LFWH AOO with selected control rod run-in (SCRRI) as well as the LFWH AOO just before the SCRRI setpoint. The staff also requested the applicant clarify the basis for the value of the FW temperature decrease (50 delta-ºC; PSAR Table 15.5-3, "Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses"). In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant clarified that FW temperature reduction following an anticipated single failure loss of FW heating not being greater than 50 delta-°C is a performance requirement established for the FW system. Based on these responses, the staff has determined that the applicant adequately evaluated the limiting LFWH temperature decrease for the preliminary design.

The applicant credits the DL2-27 Selected Control Rod Run-In on Feedwater temperature decrease, DL2-02 Maintain Target Level, and DL2-01 Maintain Target Pressure Defense Line 2 functions to mitigate the LFWH scenario. These functions are supported by non-safety-related SC3 SSCs. The core power increases due to the increase in core inlet subcooling from the LFWH event. The DL2-02 function initially compensates by lowering the FW flow rate (per

PSAR Table 15.5-5, the DL2-02 function depends on core power, reactor water level, steam flow, steam flow enthalpy, and FW enthalpy). DL2-27 initiates based on the decrease in FW temperature and inserts negative reactivity. After the initial increase in power due to the FW temperature decrease, the power decreases with decaying oscillatory behavior to a new steady-state value. FW flow, steam flow, reactor level, and other system parameters also converge to a new steady value following a period of decaying unsteady behavior. The applicant performed sensitivity studies on maximum FW pump flow, initial FW temperature, and FW controller settings. The applicant stated that the studies demonstrated no significant change in the event sequence or results. The minimum critical power ratio (MCPR) calculated in PSAR Section 15.5.3.1, "Decrease in Core Coolant Temperature AOO," is greater than the safety limit (a lower limit). As discussed in Section 15.1.3.6 of this report, **[[**

]. The analysis presented in PSAR Section 15.5.3.1, "Decrease in Core Coolant Temperature AOO," credits DL2 functions and **[[** **]]**.

With respect to PDC 26 compliance, in PSAR Section 7.3.3.2, "System Design Bases and Associated Safety Functions," the applicant stated that SCRRI allows up to eight control rods to be inserted. In PSAR Section 15.5.3.1.1, "Loss of Feedwater Heating AOO," the applicant stated that only four SCRRI rods were assumed in the LFWH AOO. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant stated that cycle-specific SCRRI rod selection will be made with consideration for a stuck rod, and a stuck SCRRI rod will be assumed in the FSAR LFWH AOO analysis. In the same letter, the applicant clarified the control reactivity assumptions related to GDC 27. The applicant clarified that the analysis did not model a stuck rod during reactor scram. The applicant presented a comparison of ΔCPR/ICPR for the BWRX-300 analyses described in PSAR Sections 15.5.4.2.1 (Turbine Trip AOO) and 15.5.5.2.2 (Turbine Trip with Half Scram DEC), which shows very little effect even with half of the control rods with the highest rod worth failing to insert. With respect to conservative scram characteristics, the applicant provided justification for the ways in which the PSAR Table 15.5-4, "CRD Scram Time," is conservative as an adequate alternative to the 0.8 multiplier on scram rate considered acceptable for use for BWRs by SRP Section 15.1.1 – 15.1.4.

The staff concludes that the preliminary analysis related to the decrease in core coolant temperature event category presented in the PSAR is sufficient to evaluate the design and performance of SSCs and assess the risk to public health and safety resulting from operation of the facility, consistent with the requirements in 10 CFR 50.34(a). Therefore, this preliminary analysis is sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design and supports issuance of a CP in accordance with 10 CFR 50.35. Further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.5.3.3.2 Increase in Reactor Pressure (AOO) Event Category

The applicant included four BL-AOOs in the reactor pressure increase (PI) category in PSAR Section 15.5.3.2: generator load rejection (turbine trip), closure of one main steam reactor isolation valve, loss of condenser vacuum, and loss of preferred power. The AOO pressurization events are analyzed as BL-AOO, which is acceptable for preliminary scoping calculations. For the FSAR, the events will need to be analyzed using the limiting conditions in the operating band for RPV pressure, FW temperature and core flow vs. any given core thermal power established in the PSAR and described in Section 4.4.3.3 of this report and assume applicable single failures. During the regulatory audit, the applicant clarified the ways in which the PSAR Table 15.5-4, “CRD Scram Time,” is conservative as an adequate alternative to the 0.8 multiplier on scram rate considered acceptable for use for BWRs by SRP Section 15.2.1–15.2.5, “Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)” and SRP Section 15.2.6, “Loss of Nonemergency ac Power to the Station Auxiliaries.” The applicant included the LOPP in the PI category, which ordinarily would be reviewed under a different event category (i.e., SRP Section 15.2.6 regarding loss of nonemergency power). While BWRX-300 does not have recirculation pumps to trip on LOPP, it does have FW pumps and if those lose power, a reactor shutdown would be required. LOPP ends up being a PI event because the loss of power results in the generator output breakers opening and the turbine control valves (TCVs) fast closure. The initial pressure increase due to the TCV fast close is limited by the anticipatory scram (DL2) and turbine bypass valves (TBVs) opening; the TBVs subsequently reclose and the pressure increase resumes. Either way, the SRP acceptance criteria of SRP Section 15.2.1–15.2.5 encompass the SRP acceptance criteria of SRP Section 15.2.6 and there is no need to uniquely apply the acceptance criteria of SRP Section 15.2.6 to the LOPP AOO event.

The MCPR and the peak pressure are the two key figures of merit for the PI AOO events; see PSAR Table 15.3-1, “Anticipated Operational Occurrence Deterministic Safety Analysis Acceptance Criteria,” for the full list of AOO acceptance criteria. According to PSAR Table 15.7-1, “Results Summary of AOO Events,” the Closure of One Main Steam Reactor Isolation Valve (1MSRIVC) AOO is limiting for ΔCPR/ICPR simulated thermal power (STP) and maximum neutron flux while load rejection-turbine trip (LR-TT), loss of condenser vacuum, and LOPP all result in the same limiting result for RPV pressure increase. The 1MSRIVC scenario is limiting for the MCPR SAFDL; see Section 4.4 of this report for a description of this SAFDL. The event sequence can be read in PSAR Section 15.5.3.2.2. Staff notes that the preliminary event sequence includes “second MSRIV in the second steam line closes on leak detection indication,” which will need to be revisited at the OL licensing stage, at which point the leak detection indication system has been designed; see letter dated April 1, 2026 (TVA 2026-TN13164, nonproprietary; ML26091A348, proprietary). This letter explains what “Line Break Indication” (LBI) means, which is the originator of a DL3 safety function to be credited for LOCAs. The letter informs the staff that the LBI system is being designed, and instead of simulating what such a system physically does, the PSAR CN-DSA used bounding timing delays on isolation signals and valve closure times. This relates to the limiting PI AOO event because it means that once the LBI system is designed and its performance parameters known, the 1MSRIVC event will need to be checked to see if the peak pressure results are different using the actual LBI system performance, especially given that this is currently the limiting AOO for ΔCPR/iCPR, which is to be used to compute operating limit MCPR (OLMCPR—see SE Section 4.4 for further information about calculating this operating limit).

PSAR Section 15.5.3.2.2 states that the following DL2 functions are credited for 1MSRIVC AOO: maintain target (reactor vessel water) level (DL2-02), anticipatory hydraulic scram on main steam reactor isolation valve (MSRIV)/main steam containment isolation valve (MSCIV) valve position (DL2-21), and passive actuation of one ICS train (DL2-31). The mechanical over-power and thermal over-power limits[1] are not exceeded by the AOO, and the event is not limiting for linear heat generation rate (LHGR); therefore, there is no fuel failure for the event.

[1] See PSAR Table 15.3-1, “Anticipated Operational Occurrence Deterministic Safety Analysis Acceptance Criteria,” top two rows in the “Fuel Rod” fission product barrier category and Section 4.2.3.3 of this report.

The AOO pressurization events have two means of removing the decay heat. They can remove decay heat through the turbine bypass system (DL2-09) and through the ICS (DL2-31). One train of ICS activates when the DL2-31 pressure setpoint is reached. The DL2-31 high reactor pressure setpoint for activation of the ICS has not yet been determined and the simulations are terminated before the ICS would activate for all AOO calculations in this category. The Generator Load Rejection or Turbine Trip events remove heat through the turbine bypass system. The 1MSRIVC and the LOPP AOO require heat removal through the ICS after a short period of turbine bypass flow heat removal until that path is lost. The peak pressure reached will depend on the value chosen for the DL2-31 pressure setpoint that actuates a single train of ICS. The Loss of Condenser Vacuum AOO starts out removing heat through the turbine bypass system. Heat removal then transitions from the condenser to the ICS after the turbine bypass valve is closed due to high condenser pressure (DL2-14) and the DL2-31 high reactor pressure setpoint is reached. The transition to ICS heat removal occurs long after the MCPR is reached in the calculation and would not affect the MCPR.

The calculations are primarily concerned with the MCPR and not with the peak pressure since the pressurizations are not a challenge to the design limit (though the DL2-31 high pressure setpoint that would determine the pressure limit has not been established). The MCPR is acceptable in all calculations. The peak pressures in the AOO calculations are bounded by the peak pressures in the DBA calculations (PSAR Section 15.5.4.2), which rely on the higher pressure DL3 ICS activation setpoints to limit the peak pressure. The events that remove heat through the turbine bypass system appear to be removing more heat than the rated system capacity. That is because the rated system capacity is based on normal operating pressure. The flow through the turbine bypass valve is higher than the rated flow because of the increased pressure. The condenser will have enough capacity to handle the increased flow since the system is designed to handle a turbine trip using the turbine bypass system without activating the ICS. As part of the OLA, the applicant will need to provide analysis of an AOO pressurization transient that extends through actuation of ICS to establish a design basis for the currently unspecified DL2-31 trip setpoint.

Regarding PSAR Table 1.9-15, “Conformance with NUREG-800 (Chapter 15 Transient and Accident Analysis),” sheets 5 through 8, the applicant stated their position on either conformance or alternative approaches to the acceptance criteria in SRP Section 15.2.1–15.2.5. For the PI AOOs (including LOPP), the applicant’s positions regarding conformance with SRP Section 15.2.1–15.2.5 acceptance criteria and associated staff evaluation are provided below:

- Acceptance criteria II.1.C and II.2.D (use of an alternate approach to conformance with guidance in RG 1.105 (NRC 2021-TN13305), “Instrument Spans and Setpoints”)—The applicant specified that an alternate approach given by IEC 61888:2002 was used. GVH states setpoints and the method for determining them will be provided with technical specifications at the FSAR stage and preliminary setpoints were used in the PSAR. The use of this alternate will be reviewed by staff at the OL licensing phase during the review of the FSAR.
- Acceptance criteria II.1.D and II.2.F (single failure, including applicability to passive system check valves)—For electrical equipment, the applicant specified an alternate approach given by IEEE Standard 379-2014 and IEEE Standard 603-2018, along with an alternate PDC 17. See Section 15.1.3.7 and Chapter 8 of this report for discussion of PDC 17. Consistent with the BL-AOO approach, a non-electrical single failure was not applied for the PI AOOs. The staff evaluated this general approach in Section 15.1.3 of this report.

- Acceptance criterion II.3.A (102 percent rated thermal power)—PSAR Table 1.9-15 states an alternative to an initial condition of 102 percent rated thermal power is used, consistent with use of nominal initial conditions in the BL-AOO analysis

The staff concludes that the BL-AOO reactor pressure increase analysis provided in the PSAR is sufficient to evaluate the design and performance of SCCs and assess the risk to public health and safety resulting from operation of the facility, consistent with the requirements in 10 CFR 50.34(a). Therefore, this preliminary analysis is sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design and supports issuance of a CP in accordance with 10 CFR 50.35. Further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.5.3.3.3 Decrease in Reactor Coolant Inventory (AOO) Event Category

PSAR Section 15.5.3.3, “Decrease in Reactor Coolant Inventory AOO,” presents the nominal, expected response to the trip of the active FW pump (Loss of Feedwater Flow [LOFW]). The analysis assumes that the non-safety-related standby FW pump starts after the loss of the operating FW pump (DL2-25). The calculated results indicate that the plant experiences a relatively minor transient response where plant safety is not compromised and plant returns to a normal operating condition. In response to Request for Additional Information (RAI) 15.5-2, (ML26111A161, non-proprietary; ML26111A162, proprietary), the applicant indicated that the elevation of reactor water level L3, which is a hydraulic scram setpoint, was set to allow scram avoidance margin in a single FW pump trip transient. That is, with this scram setpoint on reactor water level L3, the BWR will continue generating steam with a lowering water level, from a nominal water level L5 decreasing toward L3, but not reaching the L3 scram setpoint in the time it takes for the parallel, idle FW pump to start and come up to speed.

The AOO analysis provides useful information about the expected plant response during a loss of the running MF pump but does not assess a complete loss of feedwater low. Because a complete LOFW flow is considered the proper AOO, the staff assessed the complete LOFW DBA, described in PSAR Section 15.5.4.4.1. The LOFW DBA does not credit start of the standby FW pump and the transient pressure remains below the 110 percent design pressure requirement. In addition, in a letter dated March 2, 2026 (TVA 2026-TN13052), the applicant states that the LOFW DBA evaluated in PSAR Section 15.5.4.4.1 does not challenge SAFDLs, which is required for AOOs under the applicable GDCs (e.g., GDC 10, GDC 34). In the LOFW DBA results presented in the PSAR, the fuel average temperature and PCT decrease following the LOFW, which qualitatively supports the applicant’s assertion that SAFDLs are not challenged. The simulation is terminated around 300 seconds, where the major plot parameters are relatively steady. The applicant anticipates that the ICS will actuate on high RPV pressure after this time. In a letter dated March 2, 2026 (TVA 2026-TN13052), the applicant states all of the pressure increase DBAs (presented in PSAR Section 15.5.4) have similar heat loads when ICS is initiated and show that ICS is capable of removing residual heat and keeping the peak pressure to values below the design limit, even when only one of three trains is credited. The staff anticipates that the non-LOCA decrease in coolant inventory scenarios will have similar long-term decay heat loads. The staff concludes that the preliminary analysis related to the inventory reduction event category for the LOFW DBA (PSAR Section 15.5.4.4.1), and the supplemental information provided in the letter dated March 2, 2026 (TVA 2026-TN13052), is sufficient to evaluate the design and performance of SCCs and assess the risk to public health and safety resulting from operation of the facility consistent with the requirements in 10 CFR 50.34(a). Therefore, this preliminary analysis is sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design and supports issuance of a CP in accordance with 10 CFR 50.35. Further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.5.3.3.4 Increase in Reactor Coolant Inventory (AOO) Event Category

The applicant describes the increase in reactor coolant inventory AOO in PSAR Section 15.5.3.4. The applicant only selected one event in the II BL-AOO category “Inadvertent Isolation Condenser Initiating – One Train” for inclusion in the PSAR. The actuation of a train of the ICS dumps a small volume of subcooled liquid directly in the reactor vessel chimney. The injection of liquid into this region can impact core reactivity because the primary impact is due to the effect of the perturbation in the chimney density on the core flow and pressure. The applicant has specified startup testing in PSAR Table 14.1-1, “Objective of First-of-a-Kind Testing,” which has experiments with ICS condensate return to the chimney.

The staff notes this AOO has the largest neutron flux increase. However, the event is not limiting for AOO derived acceptance criteria. As a CRN-1 BWRX-300 PSAR AOO, this event credits non-safety related DL2 functions “maintain target level” and “maintain target pressure.” This AOO does not result in a reactor shutdown. Note that on PSAR plots showing a “% rated” ordinate for isolation condenser/condensate (IC) flow (i.e., PSAR Figures 15.5-40 and 15.5-42), this is a measurement of IC flow as a fraction of total reactor steam flow rate, added up for the number of activated IC loops.

The applicant states in PSAR Table 1.9-15 that the acceptance criteria, with the exception of including 2 percent uncertainty on reactor power and a 0.8 multiple for scram reactivity insertion rate, conform to SRP Section 15.5.1–15.5-2, Revision 2, “Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory.” Many concerns for inadvertent ECCS that are described by this portion of the LWR SRP are not applicable to the ECCS system of CRN-1 BWRX-300; only a small amount of liquid coolant mass is being recombined with the bulk RPV coolant mass by actuation of the ECCS. This coolant mass was held up in the ICS condensate line(s) and already within the RCPB. The calculation shows the impacts of the transient on CPR and RPV pressure are small, as shown in PSAR Table 15.7-1, “Results Summary of AOO Events.”

Considering SRP Section 15.5.1–15.5-2, the applicant did not use 102 percent rated thermal power. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant explained the ways in which the PSAR Table 15.5-4, “CRD Scram Time,” is conservative as an adequate alternative to the 0.8 multiplier on scram rate considered acceptable for use for BWRs by SRP Section 15.5.1–15.5.2. With regard to instrument setpoints, the applicant has used IEC 61888:2002 as an alternate to RG 1.105 (NRC 2021-TN13305), “Setpoints for Safety-Related Instrumentation.” Preliminary instrument setpoints were used for the PSAR analysis, but in a letter dated March 2, 2026 (TVA 2026-TN13052), GVH states that, “[t]he setpoints…are determined with the setpoint methodology defined in IEC 61888…using the final analytical limits from the plant safety analyses…;” which staff notes is a valid position taken by the applicant and provides reasonable assurance that the final setpoints and the method for determining them will be provided in the FSAR. As noted previously in this section, the FSAR should use initial conditions that represent expected variation within the operating band for RPV pressure, FW temperature and core flow vs any given core thermal power established in the PSAR and described in Section 4.4.3.3 of this report. However, the staff concludes that this AOO analysis is sufficient to evaluate the design and performance of

SCCs and assess the risk to public health and safety resulting from operation of the facility, consistent with the requirements in 10 CFR 50.34(a). Therefore, this preliminary analysis is sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design and supports issuance of a CP in accordance with 10 CFR 50.35. Further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.5.3.4 Conclusion

The staff concludes that the preliminary plant design with regard to AOOs is consistent with the requirements in 10 CFR 50.34(a) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

15.5.4 Analysis of Design Basis Accidents

15.5.4.1 Introduction

This section presents the DSA of DBAs for the BWRX-300 as described in the CRN CPA. The purpose of the DBA analysis is to evaluate the plant's response to a spectrum of postulated accident conditions—specifically those that are less frequent but more severe than AOOs—and to demonstrate that the facility design provides adequate protection of public health and safety in accordance with NRC requirements.

The DBA analyses presented herein are performed in accordance with the BWRX-300 Safety Strategy and are structured to meet the requirements of 10 CFR 50.34, 10 CFR 50.46, and 10 CFR Part 50 (TN249), Appendix A (General Design Criteria). The analyses address a comprehensive range of PIEs identified through systematic fault evaluation and include accidents such as large and small break LOCAs, main steam and FW line breaks, and other limiting scenarios.

15.5.4.2 Regulatory Evaluation

The staff based its review on the relevant requirements of the following regulations and guidance:

- 10 CFR 50.46 (TN249)
- 10 CFR 50.34(a)(4)
- 10 CFR 50.35
- GDCs 10, 13, 15, 20, 25, 27, 31, 34, and 35, as identified in Section 15.1.2 of this report

15.5.4.3 Technical Evaluation

In reviewing PSAR Section 15.5.4, the staff has concluded that the DBA analysis presented in the PSAR, and described in the sections below, is sufficient to evaluate the design and performance of SCCs and assess the risk to public health and safety resulting from operation of the facility in accordance with 10 CFR 50.34(a). While reviewing, the staff observed that some DBAs were not consistently analyzed using only safety related SSCs for mitigation, consistent with the definition of safety related in 10 CFR 50.2. Furthermore, as discussed in 15.1.3 of this report, the DBA analyses presented in the PSAR did not systematically evaluate initial conditions, power availability, single failure requirements, and safe shutdown fuel performance criteria. However, 10 CFR 50.35 permits issuance of a CP when, among other things, sufficient preliminary information is provided in the CPA and the NRC staff determines that further information required to complete the safety analysis, which can reasonably be left for later consideration, can be supplied in the FSAR. The staff's evaluation and conclusions with respect to the DBA analyses presented in the PSAR are further discussed in the subsections below.

Staff notes that both the BL-AOOs and CN-DBAs point to PSAR Table 15.5-3, "Input Parameters and Initial Conditions and Assumptions used in Non-LOCA Analyses," which has a single set of conditions for plant parameters[2], initial conditions[3], and biases and uncertainties[4]. This means that the AOOs and DBAs were either both done conservatively or as baseline (nominal) simulations. In a letter dated March 2, 2026 (TVA 2026-TN13052), the staff learned that not all of the CN-DBAs were analyzed with fully conservative settings for plant parameters, initial conditions, biases and uncertainties.

In the March 2, 2026, letter, the applicant states that, "ongoing work for the FSAR implements a more rigorous process to address uncertainties in the CN-DSA…the process being implemented for FSAR CN-DSA addresses uncertainties in initial conditions and in model parameters for the limiting event in each fault group for non-LOCA events." The letter goes on to state "statistical methodology (as described in NEDC-34043, Revision 1, Sections 3.6.3 and 3.6.4) will be performed for each of these limiting events…the preliminary results of the in-progress CN-DSA being performed for FSAR thus far demonstrate that the graded approach utilized for the PSAR phase was effective in identifying the limiting event and assessing the uncertainty." Staff will review the analyses in the FSAR submitted as part of an OLA to confirm it is performed in accordance with the letter dated March 2, 2026, using conservative DSA and statistical quantification of uncertainty.

15.5.4.3.1 Decrease in Core Coolant Temperature (DBA) Event Category

The decrease in core coolant temperature DBA event category is discussed in PSAR Section 15.5.4.1. The applicant identified a loss of all FW heaters as the event that will result in the largest reduction in FW temperature and that it is limiting for PCT. The applicant states that ICS, CRD, CUW, and SDC can only reduce the coolant temperature a small fraction relative to FW heating related PIEs. Based on the preliminary design information, the staff agrees that the loss of all FW heating scenario will result in a greater decrease in core coolant temperature than the scenarios involving CRD, CUW, and SDC. Inadvertent actuation of all ICS is evaluated in the increase in reactor coolant inventory event category. In PSAR Section 15.2.4, the applicant stated that (1) full power conditions are expected to result in the most significant challenge to the fission product barriers and (2) the identification of the most limiting scenarios under all

[2] In this case, a *plant parameter* is things such as protection system setpoints and valve capacities that influence the characteristics of the transient response, but which do not (when properly prescribed) have an impact on steady-state operation. (GVH's terminology, NEDE-32906P-A, Revision 3).

[3] In this case, an *initial condition* is key plant inputs that determine the overall steady-state nuclear and hydraulic conditions prior to the transient. These are inputs that are essential to determining that the steady-state condition of the plant has been established. (GVH's terminology, NEDE-32906P-A, Revision 3).

[4] In this case, *biases and uncertainties* are indicated by data comparisons between separate effects test data and TRACG calculations performed with the best-estimate version of the code and are used to establish probability density functions (PDFs) for TRACG parameters and correlations. Biases are compensated by appropriate choice of the mean value of the PIRT multiplier and uncertainties are accommodated by choosing PDFs to represent the standard deviation of the data comparisons. (GVH's terminology, NEDE-32906P-A, Revision 3).

conditions (e.g., low power) cannot confidently be performed until the design is complete and DSA of the final design is performed, which cannot practicably be achieved until the FSAR phase. The staff will confirm that the limiting scenario was identified once final design information is available in the FSAR. The staff did not review for potential human errors that could produce limiting PIEs or increase the severity of the LFWH scenario presented. In PSAR Section 15.1.2.2, the applicant clarified that the inputs needed to complete the HFEA are incomplete at the PSAR stage and will be complete by the FSAR. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant clarified that FW temperature reduction time constant assumed in the analysis is a performance requirement (performance requirement is intended to bound actual performance).

The applicant credits the following DL3 safety-related functions for the loss of all FW heating scenario: DL3-05–hydraulic scram on high STP, DL3-23–FW isolation on high RPV water level, DL3-17–MSRIV/MSCIV isolation on low RPV pressure, and DL3-12–ICS train 2 initiation on high RPV pressure. The applicant assumes the reactor level control (RLC) and reactor pressure control (RPC), which are DL2 functions, fail as-is. The SC3 I&C failure mechanisms will be reviewed during the FSAR, and the outcome of that review will determine the adequacy of DL2 functions being assumed to fail as-is rather than high or low. The applicant states that the first ICS train fails to actuate (assumed single failure). The ICS is credited for the long term cooling period; however, the simulation is terminated prior to the setpoint. In a letter dated March 2, 2026 (TVA 2026-TN13052), the applicant states all of the pressure increase DBAs (presented in PSAR Section 15.5.4) have similar heat loads when ICS is initiated and show that ICS is capable of removing residual heat and keeping the peak pressure to values below the design limit, even when only one of three trains is credited. The staff anticipates that the decrease in coolant temperature DBA will also have similar long-term decay heat loads.

In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant clarified the control reactivity assumptions related to GDC 27. The applicant clarified that the analysis did not model a stuck rod during reactor scram. In this response, the applicant presented a comparison of ΔCPR/ICPR for the BWRX-300 analyses described in PSAR Sections 15.5.4.2.1 (Turbine Trip AOO) and 15.5.5.2.2 (Turbine Trip with Half Scram DEC), which shows very little effect even with half of the control rods with the highest rod worth failing to insert. With respect to conservative scram characteristics, the applicant provided justification for the ways in which PSAR Table 15.5-4, “CRD Scram Time,” is conservative and an adequate alternative to the 0.8 multiplier on scram rate is considered acceptable for use in various sections of SRP Chapter 15 (e.g., Section 15.1.1-15.1.4).

For the loss of all FW heating DBA, calculated RPV pressure remains below the 120 percent design pressure limit and the fuel PCT is shown to never increase above the initial value, prior to the termination of the event at 400 seconds (PSAR Figure 15.5-51). The fact that PCT did not exceed the initial value indicates the fuel cooling was adequate and the fuel cooling DBA acceptance criteria were intact. The staff concludes that the preliminary analysis related to the decrease in core coolant temperature event category presented in the PSAR is sufficient to evaluate the design and performance of SCCs and assess the risk to public health and safety resulting from operation of the facility, consistent with the requirements in 10 CFR 50.34(a). Therefore, this preliminary analysis is sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design and supports issuance of a CP in accordance with 10 CFR 50.35. Further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.5.4.3.2 Increase in Reactor Pressure (DBA) Event Category

The applicant describes the increase in reactor pressure DBA event category in PSAR Section 15.5.4.2. The applicant included four PI CN-DBAs in the PSAR: generator load rejection or turbine trip (LR-TT), LOPP, RPV pressure downscale, and closure of all MSRIVs and FW isolation valves. For understanding in this PI DBA event category, a “pressure downscale” event is a failure of the pressure regulator(s) or RPC system such that an erroneous signal representing a demand for pressure increase persists. The event results in slow closure of the TCVs (as opposed to fast-close), potentially causing full closure of the TCVs as well as inhibition of the TBVs, leading to increasing reactor power and pressure with an eventual scram on high neutron flux.

Staff notes that the PSAR Table 1.9-15, “Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis),” addresses SRP sections associated with the PI events described in PSAR Section 15.5.4.3.2, but specific SRP sections consider these events to be AOOs rather than DBAs. This is a consequence of the event categorization process used for the application and discussed in Section 15.1.3.2 of this report; consequently, there is no equivalent SRP section for these DBA events. However, the turbine trip, main steam isolation valve closure, and steam pressure regulator failures events are addressed in SRP Section 15.2.1-15.2.5, and the loss of preferred power is addressed in SRP Section 15.2.6. Table 1.9-15 indicates general conformance with SRP 15.2.1-15.2.5 and SRP 15.2.6 but identifies several alternate approaches, including how setpoints, single failure, and certain input parameters were addressed. Therefore, the staff used the guidance in these SRP sections to support the review of PI DBAs.

By far the most significant PI DBA is the LR-TT. The applicant also noted, in PSAR Section 15.5.1.1.2, that the event is considered medium margin, indicating “method uncertainty is addressed by biasing key important phenomena in a conservative direction (typically one or two sigma). Input parameters such as power, pressure, level, or temperature are based on using the most limiting normal operating values.”

As can be seen on the PSAR plot of neutron flux (Figure 15.5-52) or read from the table with the simulation results (Table 15.7-2), the maximum neutron flux at the time of trip on DL3-04, “hydraulic scram on high neutron flux,” was 544 percent and occurred at 0.42 seconds after event initiation. The control rods start moving 0.2 seconds later. Two additional DL3 functions are credited: “FW isolation on high RPV water level” (DL3-23) and “ICS train 2 initiation on high RPV pressure” (DL3-12). The staff notes that there was an assumed failure of the lowest pressure actuation IC train (DL3-11), making the pressure increase event more severe. PSAR Table 15.5-14 notes that ICS injection initiated at 152 seconds. Actuation of the ICS limits the peak pressure of the event to values below the design pressure.

During the regulatory audit, in a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant informed the staff that the sudden rapid increase in cladding temperature, coupled with the power spike, that the TRACG code predicts a transition to film boiling with a brief delay to rewet. This is a different behavior than is seen in the AOO counterpart calculation and is because there is no DL3 trip that is equivalent to the DL2-08 Anticipatory Hydraulic Scram on Turbine Trip or Generator Load Rejection Demand. No code qualification or assessment for transient heat-ups in non-LOCA conditions was provided in the PSAR or its reference documents. To simulate such a heat-up in a non-LOCA DBA event, GVH opted to use a particular combination of TRACG code options that has not been previously, generally approved. By reviewing the April 1, 2026, letter, the staff ascertained the specific code options selected and staff reviewed the evidence to support qualification of these TRACG options. Regarding the code option to calculate the transition from nucleate boiling to post-critical heat flux (CHF) heat transfer, GVH selects the General Electric critical quality - boiling length (GEXL) correlation, which is discussed in a limited fashion—but with references—in Section 4.4.2.3 of this report. The GEXL correlation is the most accurate model (i.e., it is best-estimate) in applicable conditions because it is based on bundle design specific data. This critical quality correlation used for calculating the critical power ratio has been previously approved in NEDE-24011-A, Revision 31, "General Electric Standard Application for Reactor Fuel (GESTAR II, US). When moving back down the boiling curve, the Shumway correlation without the void fraction term is used for switching from film boiling to transition boiling. The void fraction term increases the minimum stable film boiling temperature (T_{min}) compared to setting the term equal to one. GVH argues that this is the most accurate correlation with respect to trends in T_{min} vs. pressure, and it accounts for various cladding materials. Without the void enhancement term, the correlation conservatively predicts T_{min} to be lower than all of the data for zircaloy near the BWRX-300 pressure (see letter dated April 1, 2026 (TVA 2026-TN13164 [non-proprietary]; ML26091A348, [proprietary]) where transient heat-ups occur.

In the April 1, 2026, letter, GVH elaborated on conservatisms in the simulation of "PI-LR-TT_CCF-DL2_CN-DBA" beyond those stated in the PSAR. These conservatisms are described below:

- FW trip on L8 disabled: This actuation would normally occur at a lower RPV water level than the DL3-23 FW isolation on high RPV water level, therefore disabling the FW trip maximizes FW injection
- GEXL critical quality biased low to cause earlier boiling transition which increases PCT
- Biases for the following: Void coefficient **[[**

]].

GVH states that the conservative direction for all these was ascertained and then simultaneously applied to a single case to achieve a bounding result. Staff notes that the hot rod power peaking factor of **[[** **]]** but is not a value that is in the range of maximum values for fuel bundles in the BWRX-300, including uncertainties, as it is currently designed. The staff notes that while this hot rod power factor is certainly bounding, the concern is that it may give rise to simulation results which are distorted and obscure a more realistic plant response, for this event and other events that use this bounding multiplier. According to GVH, the combined conservative biasing had little impact on peak pressure because that is dominated by the ICS initiation pressure. According to GVH, the combined conservative biasing resulted in a 91 delta-°C increase to PCT over baseline.

Regarding the biasing on hot rod power as specified in PSAR Section 15.5.4.2.1, the NRC staff asked what is meant by "a" "hot rod," because, due to symmetry in the core, "the" hot rod may be a set of equally hot, hot rods rather than there being one single hottest rod in the core. As part of the regulatory audit, GVH expanded upon modeling details related to Design Basis Records held by GVH for the BWRX-300 design. Specifically, there are groups of rods within the fuel bundles and a hot rod is only a full length rod that is surrounded by full length rods. To implement a “hot rod” for a BWRX-300 TT DBA, which is different than the similarly named “hot rod model” in the TRACG theory manual (NEDE-32176P Revision 4), the peaking factor of the hot rod group in a hot bundle is biased high. When the PSAR says “the hot rod power was biased high,” it means a multiplier was distributed to the hot rod(s) in every hot bundle.

In the letter from April 1, 2026, GVH explained that NEDE-33005P-A, Revision 2, “TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6,” limited the use of the Shumway correlation to LOCA conditions. However, the use of the Shumway correlation has been approved by the NRC on a plant-by-plant basis for BWR EPU/MELLLA+ ATWS (Extended Power Uprate/Maximum Extended Load Line Limit Analysis, + [plus]), where ATWS is a beyond-design-basis, non-LOCA event and most similar to the CRN-1 BWRX-300 LR-TT DBA, as long as a partner calculation using the homogeneous nucleation model is made to serve as comparison to the calculation using the Shumway correlation. Considering the facts and the history of a gradual approach to accepting the use of the Shumway correlation, the staff accepts the code options specified in the preceding paragraphs for the LR-TT non-LOCA DBA with fuel heat-up, including the use of the Shumway correlation without a comparison calculation using other models, for the CRN-1 BWRX-300. In this decision, staff also considered the quality of the data supporting the correlation and the large, quantified margin to 2,200°F and 2,050°F (see PSAR Table 15.3-2, “Sheet 2 of 4”) even when the conservatisms and biases listed above were applied.

The effect of this DBA does not result in any challenge to the temperature or pressure transient derived acceptance criteria for the fuel, pressure vessel, or containment. While there is moderate cladding temperature increase, the peak is much less than 2,200°F and 2,050°F, although clearly the MCPR SAFDL has been violated. The PSAR states that no fuel failures are predicted to result from the transient; see Section 15.1.3 of this report for a description of the preliminary (i.e., as of yet unapproved) model for fuel damage and failure at temperatures between operating PCT and 2,200°F. This new fuel damage and failure model is to be reviewed in a future action, whether it be a topical report or at the OL phase.

As discussed in PSAR Section 15.5.1.2.1, “TRACG,” TRACG ICS modeling was qualified using the PANTHERS IC test program. PANTHERS-IC testing was performed in the 1990s and initially performed to support the design certification of the Simplified Boiling Water Reactor (SBWR) (e.g., see NRC 1998-TN13177). The ICS is required to perform at RPV pressures lower and higher than the pressures in the PANTHERS tests during some Chapter 15 DBA and DEC events. The lower pressure conditions are covered by the PANTHERS passive containment cooling system (PCCS) testing, which used a heat exchanger similar to the ICS heat exchanger but used in lower pressure conditions for the SBWR and ESBWR containments. The high-pressure conditions (>8.1 megapascals [MPa]) calculated in the DBA and DEC Chapter 15 increase in pressure events do not have test condition coverage from PANTHERS or other tests. It is possible that the outside of the condenser tubes could exceed the onset of CHF under those conditions and degrade the heat transfer performance of the ICS. If that were to happen, the Chapter 15 calculations would underestimate the peak pressure in the system. GVH has provided information showing that CHF is conservatively modelled using Griffith’s modification to the Zuber pool boiling model (see NUREG/CR-1559, “Transient Critical Heat Flux and Blowdown Heat-Transfer Studies,” (ML070610294) and that there is adequate margin to CHF in the DBA and DEC calculations in Chapter 15 where the pressures in the calculation exceed the pressures of the PANTHERS test data.

Because the PSAR selected LR-TT as the limiting DBA in the pressure increase event category, the staff reviewed the applicant’s conformance with SRP Section 15.2.1–15.2.5, as described in

PSAR Table 1.9-15 and with the recognition that the SRP is intended to support the review of an AOO rather than a DBA. The staff concluded the following:

- Criterion II.1.A is met as the applicant identified the limiting loss of heat sink event.
- Criterion II.1.B is satisfied for maintaining reactor pressure below 110 percent of design pressure, but there is only a preliminary result for the fuel damage aspect. The staff has not approved a model for calculating fuel damage below 2,200°F. See Section 15.1.3 of this report for a description of the preliminary model for fuel damage and failure at temperatures between operating PCT and 2,200°F.
- Criteria II.1.C, GVH states setpoints will come with technical specifications at the FSAR stage.
- Criterion II.1.D, although RG 1.206 (NRC 2018-TN6192) does not apply to a 10 CFR Part 50 CPA, single failure aspects of the design are described under acceptance criteria II.2.D, II.2.E, and II.2.F, as discussed below.
- Criterion II.2.A is met because the applicant demonstrated that the reactor pressure is maintained below 110 percent of design pressure as is shown in PSAR Table 15.7-2.
- Criterion II.2.B, which requires the SAFDLs be met, was violated. However, this is a DBA and not an AOO, so this is permissible.
- Criterion II.2.C is met, as the event should not result in an aggravated plant condition without other faults occurring.
- Criteria II.2.D, II.E, and II.F are not fully met. Final setpoints will not be established until the FSAR is developed. For RG 1.53 (NRC 2003-TN13228), “Application of the Single-Failure Criterion to Safety Systems,” the applicant specified an alternate approach given by IEEE Standard 379-2014 and IEEE Standard 603-2018, along with an alternate PDC 17. See Chapter 15.1.3.7 and Chapter 8 of this report for discussion of PDC 17. Considering non-electrical single failures, a single failure of the lowest pressure actuating IC loop was applied; however, it was not clear in PSAR documentation if this actually the limiting single failure.
- Criterion II.3.A and II.3.B are met; **[[**]] power was used as an initial **[[** **]]**. Regarding II.3.B, the applicant appropriately justified their approach for use of an alternative to a factor of 0.8 multiplier on the scram reactivity insertion rate.
- Criterion II.3.C, which specifies selecting a limiting core burnup as it would apply to a BWR, has been met. This is evidenced by the statement, “[t]he event is run at BOC, MOC, and EOR cycle exposure conditions,” in PSAR Section 15.5.4.2.1.
- Criterion II.3.D, it appears the ICS condensate return valves opening times conservatively are assumed to be at their maximum values, but according to the applicant, setpoints are not final until the FSAR stage.

For the FSAR, the events will need to be analyzed along the operating band for RPV pressure, FW temperature, and core flow versus any given core thermal power established in PSAR Section 4.4, and also using a limiting peaking factor that accounts for uncertainties as done in standard reload analysis; see section 4.4.3.3 of this report for further information.

Based on the evaluation of the specific SRP criteria discussed in the preceding paragraphs, the staff concludes that the analysis for this DBA event category is sufficient to evaluate the design and performance of SCCs and assess the risk to public health and safety resulting from operation of the facility, consistent with the requirements in 10 CFR 50.34(a). Therefore, this preliminary analysis is sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design and supports issuance of a CP in accordance with 10 CFR 50.35. Further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.5.4.3.3 Increase in Reactor Coolant Inventory (DBA) Event Category

The applicant describes the increase in reactor coolant inventory DBA event category in PSAR Section 15.5.4.3. The applicant included two II CN-DBAs in the PSAR: Feedwater flow increase (FWFI) for all pumps and inadvertent actuation of the ICS (all trains). As was explained above in the "Increase in Reactor Pressure (DBA)" subsection, even though the PSAR denotes the "Increase in Reactor Coolant Inventory (DBA)" events as "CN," the simulations were not strictly conservative; the staff will review the final analysis described in the FSAR at the OL licensing phase. The FWFI DBA (PSAR Section 15.5.4.3.1) assumes both FW pumps fail instantly to maximum FW flow, causing RPV water level increase. Four DL3 functions are credited: hydraulic scram on high STP (DL3-05), FW isolation on high RPV water level (DL3-23), MSRIV/MSCIV isolation on low RPV pressure (DL3-17), ICS train 2 initiation on high RPV pressure (DL3-12). The transient is not run out to the point where ICS initiates.

The inadvertent actuation of all trains of ICS (PSAR Section 15.4.3.2) releases cold water into the chimney region of the RPV and an increase in the amount of heat removed from the system. This leads to a decrease in pressure with the cold water in the chimney condensing steam and this leads to a decrease in power due to the increase in the core void fraction, which results from a decrease in pressure. The power and pressure continue to drop until the end of the calculation. While the inadvertent actuation of all trains of ICS is not limiting with respect to RPV pressure, the PSAR plot shows multiple instances of negative flow for the ICS. GVH claims this reverse flow is caused by the rapid 1 second IC condensate return valve opening time. That is feasible for the initial valve opening but the largest negative IC flow in duration and magnitude occurs at ~50 seconds after the start of the transient and is well removed from the IC isolation valve opening. Note that on PSAR plots (PSAR Figures 15.5-82 and 15.5-84) showing a "% rated" ordinate for IC flow, this is a measurement of IC flow as a fraction of total reactor steam flow rate, added up for the number of activated IC loops.

The FWFI is more limiting for each DBA-derived acceptance criteria, but this is significantly less limiting than the LR-TT PI DBA. The staff needs further design details to determine if flow reversal in the ICS could result in inadvertent ICS isolation. This is discussed in further detail in Section 15.5.4.3.6 of this report under the topic of "Decrease in Reactor Coolant Inventory (DBA) Event Category (LOCA), ECCS Performance."

For the FSAR, the event(s) should be analyzed along the operating band for RPV pressure, FW temperature and core flow vs. any given core thermal power established in the current PSAR and described in Section 4.4.3.3 of this report. However, the staff concluded that the analysis for this DBA event category is consistent with the requirements in 10 CFR 50.34(a) and sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design. Therefore, this preliminary analysis supports issuance of a CP in accordance with 10 CFR 50.35 and further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.5.4.3.4 Decrease in Reactor Coolant Inventory (DBA) Event Category (Non-LOCA)

The applicant described the non-LOCA decrease in reactor coolant inventory DBA event category in PSAR Section 15.5.4.4. The analysis in this event category is limited to non-LOCA; LOCA DBAs are evaluated in a subsections 15.5.4.3.5 and 15.5.4.3.6 of this report. The applicant identified loss of all FW flow, opening of all TCVs and TBVs, and RPV pressure control open as PIEs relevant to this event category. During the regulatory audit (NRC 2025-TN13059), the staff determined that RPV pressure control open and demand to open all TCVs and TBVs were the same event sequence. The selected bounding event is the loss of FW flow event, and the RPV pressure control open event is also evaluated and considered potentially limiting. In PSAR Section 15.2.4, the applicant stated that (1) full power conditions are expected to result in the most significant challenge to the fission product barriers and (2) identification of the most limiting scenarios under all conditions (e.g., low power) cannot confidently be performed until the design is complete and DSA of the final design is performed, which cannot practicably be achieved until the FSAR phase. The staff will confirm that the limiting scenario has been identified once final design information is available in the FSAR. Based on the preliminary design information provided in PSAR Figure 5.1-5 and in a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary, ML26091A348, proprietary), the staff agrees that inventory reduction via LOFW or RPV pressure control open are limiting compared to non-LOCA scenarios involving CUW or SDC.

The applicant states that the LOFW event could be initiated by FW isolation valve closure DL4a failure but is not selected because it is less severe than LOFW. In addition, in the regulatory audit, the applicant stated that FW isolation via DL4a failure and DL2 failure is considered a DEC based on the event sequence frequency. As discussed in Section 15.1.3.1 of this report, the event category is based on the initiating event frequency; a DBA PIE remains a DBA when non-safety-related SSCs (e.g., those performing DL2 functions) are not credited to demonstrate compliance with the applicable regulations (e.g., 10 CFR 50.2). In a letter dated February 4, 2026 (TVA 2026-TN13093), the applicant clarified that the LOFW is considered more limiting than FW isolation because of the potential for reverse flow through the un-isolated FW line following loss of all FW flow. The applicant also clarified that the DL3-39, “FW isolation on loss of normal FW flow,” function actuates after a 30-second delay and is designed to actuate only if the SC3 FW check valves are not working as designed. In the PSAR analysis, the FW flow decreases to zero around 10 seconds. In the regulatory audit, the applicant clarified that the assumed nominal FW pump trip coast-down time constant is a design requirement. The applicant states that leakage from 10 to 30 seconds is not modeled, and the presented results show zero reverse FW flow. The applicant states that the additional inventory loss via FW check valve leakage (i.e., leakage due to DL2-43 failure) for 30 seconds would have an insignificant effect on reactor water level, which has significant margin to top of active fuel, as shown on PSAR Figure 15.5-90.

For the LOFW scenario, the initiating event is a complete loss of FW flow. The applicant credits the following DL3 safety-related functions:

- DL3-03—hydraulic scram on low RPV level
- DL3-17—MSRIV/MSCIV isolation on low RPV pressure
- DL3-14—ICS initiation on low RPV water level
- DL3-39—FW isolation on Loss of Normal FW flow

The applicant assumes the RPC, a DL2 function, fails as-is. The SC3 I&C failure mechanisms will be reviewed during the FSAR, and the outcome of that review will determine the adequacy of DL2 being assumed to fail as-is rather than high or low. Following LOFW, the level and pressure slowly decrease as inventory is lost through the steamline. The decrease in pressure corresponds to a modest increase in core average void fraction and a decrease in power. The reactor scrams on low RPV level around 30 seconds. The inventory loss is terminated with the MSRIV isolation on low RPV pressure around 85 seconds (and assuming FW isolation around 30 seconds, although this is not explicitly modeled). The applicant credits the DL3-14 (ICS initiation on low RPV water level) function at 130 seconds. All ICS trains are modeled to open. The applicant states that a single failure of one ICS train to start does not affect event mitigation. The simulation is terminated at 300 seconds, when the system parameters are relatively steady and long-term heat removal via ICS is underway.

For the RPC open scenario, the applicant assumes the event is initiated by all TCVs and TBVs fully opening with RLC failing as-is. The applicant credits the following safety-related DL3 functions: DL3-02—hydraulic scram on low RPV pressure; DL3-17—MSRIV/MSCIV isolation on low RPV pressure; DL3-23—FW isolation on high RPV water level; and DL3-12—ICS train 2 initiation on high RPV pressure. The pressure decreases monotonically from event initiation until the low RPV pressure is reached, resulting in reactor scram and MSRIV isolation initiation. After an initial decrease in level, the level increases due to the continued FW pump flow. In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant clarified the modeling assumptions for the FW heating as a function of steam flow. At around 225 seconds, the FW flow isolates on high RPV level. The applicant assumes a single-failure results in the first ICS train failing to actuate and the scenario is terminated prior to ICS initiation.

In a letter dated March 2, 2026 (TVA 2026-TN13052), the applicant states that all pressure increase DBAs (presented in PSAR Section 15.5.4) have similar heat loads when ICS is initiated and show ICS to be capable of removing residual heat and keeping the peak pressure to values below the design limit, even when only one of three trains is credited. The staff anticipates that the decrease in coolant inventory DBAs also will have similar long term decay heat loads.

In a letter dated April 1, 2026 (TVA 2026-TN13164, non-proprietary; ML26091A348, proprietary), the applicant clarified the control reactivity assumptions related to GDC 27. The applicant clarified that the analysis did not model a stuck rod during reactor scram. The applicant presented a comparison of ΔCPR/ICPR for the BWRX-300 analyses described in PSAR Sections 15.5.4.2.1 (Turbine Trip AOO) and 15.5.5.2.2 (Turbine Trip with Half Scram DEC), which show very little effect even with half of the control rods with the highest rod worth failing to insert. With respect to conservative scram characteristics, the applicant provided justification for the ways in which the Table "15.5-4 CRD Scram Time" is conservative as an adequate alternative to the 0.8 multiplier on scram rate considered acceptable for use for BWRs by SRP Section 15.2.7.
For both the LOFW and RPC open scenarios considered in the PSAR, the fuel PCT did not increase above the initial values, meaning cooling was adequate and the fuel cooling DBA acceptance criteria were met; see PSAR Figures 15.5-93 and 15.5-99. The RPV pressure remained below 120% of design pressure limit. Furthermore, the staff also notes the RPV pressure remained below the 110 percent pressure limit, which is the acceptance criterion on RCPB integrity for AOOs established in the PSAR. In a letter dated March 2, 2026 (TVA 2026-TN13052), the applicant states that the LOFW DBA evaluated in PSAR Section 15.5.4.4.1 also does not challenge SAFDLs, which would be required for AOOs under the applicable GDCs. In

the results presented in the PSAR, the fuel average temperature and PCT decrease following the LOFW, which supports the applicant's assertion regarding SAFDL integrity during this DBA.

Based on the above, the staff concludes that the preliminary analysis related to the inventory reduction event category is sufficient to evaluate the design and performance of SCCs and assess the risk to public health and safety resulting from operation of the facility, consistent with the requirements in 10 CFR 50.34(a). Therefore, this preliminary analysis is sufficient to permit the applicant to describe major features and the principle architectural and engineering criteria for the design and supports issuance of a CP in accordance with 10 CFR 50.35. Further technical and design information can reasonably be left for later consideration and will be reviewed with the FSAR at the OL licensing phase.

15.5.4.3.5 Decrease in Reactor Coolant Inventory (DBA) Event Category (LOCA)—Containment Response

The applicant described the containment response for the decrease in reactor coolant inventory DBA event category in PSAR Section 15.5.4.5. The applicant included analyses for containment response biased cases for four LOCAs in the PSAR: steamline and FW line large break LOCAs (LBLOCAs) inside containment and steam and liquid small break LOCAs (SBLOCAs) inside containment. An acceptable method to evaluate the BWRX-300 containment response to a LOCA is outlined in NUREG-0800 (NRC 2021-TN8013), SRP Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments." Although this SRP section refers to PWR dry containments, this is appropriate for the dry containment design of the BWRX-300. While the applicant's proposed BWRX-300 design does not have the traditional pressure suppression containment consisting of a drywell and a wetwell with a suppression pool, where the drywell and wetwell are connected by the vents and the vacuum breakers, theRPV is in a dry containment and the containment pressure is passively limited during a LOCA using the PCCS. In addition, the ICS can also play a role in limiting containment pressure by reducing mass and energy release into the containment from the RPV.

The PSAR LOCA evaluation uses the methodology described in the LTR, NEDC-33922P-A, "BWRX-300 Containment Evaluation Method" (GE Hitachi 2022-TN13095, non-proprietary; ML22168A013, proprietary). The methodology uses GOTHIC and TRACG codes and is based on biasing the initial conditions and modeling parameters to account for uncertainties. Based on results presented in PSAR Table 15.7-3, "Results Summary of DBA Events for LOCA," and PSAR Table 6.2-1, "Primary Containment System Key Design Parameters," the main steam pipe break inside containment resulted in the greatest challenge to the containment pressure design limit; that is, 322 kilopascals (kPa) gauge for the main steam pipe break peak pressure vs. 414 kPa gauge containment design limit (rounded numbers). Main steam and main feedwater pipe breaks inside containment have the same peak containment shell temperature and were the greatest challenges to containment shell temperature limit (134°C vs the 165.5°C design temperature). The fast isolation valve design in the BWRX-300 proposed at the CRN-1 Site is effective at limiting the peak pressure to values nearly 1 atmosphere (i.e., 92 kPa or 13.3 pounds per square inch [psi]) below the design pressure. This is a large margin compared to currently operating reactors.

Fuel cooling/ECCS performance focused LOCA analyses are covered in Section 15.5.4.3.6 of this report.

Addressing Four Limitations and Conditions on Licensing Topical Report NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," in PSAR Section 15.5.4.5

The SER for the LTR NEDC-33922P-A specifies four L&Cs that must be met when applying the methodology. PSAR Section 15.5.4.5, "Loss of Coolant Accidents Design Basis Accidents," summarizes information on meeting the four L&Cs in the applicant's CPA. PSAR Section 1.6.1 documents that NEDC-33922P-A describes the DBA containment analysis methodology and is incorporated by reference in the applicant's PSAR. Therefore, as part of the CPA, the applicant needs to implement the four L&Cs for the BWRX-300 reactor design referenced in the PSAR. The four L&Cs developed by the staff during the review of NEDC-33922P-A require demonstration that the fuel and containment integrity acceptance criteria are met for at least 72 hours for the limiting DBEs using only passive heat removal systems (i.e., ICS and PCCS). The PCCS and ICS designs are discussed in PSAR Sections 6.2.2 and 6.3, respectively. In order to perform an evaluation of the four L&Cs as addressed in PSAR Section 15.5.4.5, the staff conducted an audit of the material furnished by the applicant in the electronic reading room in response to Audit Questions A-6.2-1, A-6.2-4, and A-6.2-9, as well as the GVH report, NEDC-34349P, Revision 0, "BWRX-300 Darlington New Nuclear Project (DNNP) Joint Report on GE Hitachi's Containment Evaluation Method: Implementation of Licensing Considerations," that TVA made available for audit. The staff performed the audit to support the PSAR Chapter 6 and Chapter 15 reviews in regard to the technical details of the applicant's approach to containment safety analysis and addressing the four L&Cs.

L&C#3 and L&C#4 were developed to address the Chapter 6 safety concerns about the containment cooling and pressure mitigation through the PCCS under DBA conditions. However, L&C#3 and L&C#4 are not addressed in PSAR Chapter 6 and are instead addressed in PSAR Section 15.5.4.5 with L&C#1 and L&C#2, which are relevant to Chapter 15 reactor systems. Through the PSAR review and as verified by the audit, the staff has determined that Chapter 15 results as well as L&Cs implementation are based on the new (i.e., modified) PCCS design being referenced in the PSAR. Analyzing the new PCCS design was an important requirement for meeting L&C#4 that also affected the safety-significant reverse flow analysis from the containment to RPV for L&C#3. In the following paragraphs, the four L&Cs and the staff evaluation of PSAR Section 15.5.4.5 are addressed.

L&C #1—Isolation Condenser Radiolytic Gas Removal

The use of this Containment Evaluation Method (CEM) is limited to a BWRX-300 design that limits the total volumetric fraction of radiolytic gases in the IC lower drum to a sufficiently low level throughout a 72-hour period following the event such that condensation heat transfer in the ICs is not adversely affected and the hydrogen deflagration margin is maintained.

The applicant intends to meet L&C#1 by adding an autocatalytic recombiner to each train of the ICS system. It is stated in PSAR Section 15.5.4.5 that for L&C#1, "The analyses assume that the isolation condensers are equipped with features to maintain the radiolytic gases in the lower drum sufficiently low to prevent deflagration of combustible gases and degradation of the IC heat removal rate. The design feature to control radiolytic gas concentration will be provided in the FSAR." The design is not finalized but a preliminary design of the recombiner is shown in PSAR Figure 6.3-2. The recombiner needs to keep the radiolytic non-condensable gas concentration low enough to prevent detonation of the mixture and prevent degradation of the condensation heat transfer so that the ICS can meet its specified heat removal capacity. GVH will conduct a test program to validate the functionality of the autocatalytic recombiner. The test program is described in NEDC-34349P, which the staff audited. The staff finds the addition of an autocatalytic recombiner to the ICS to be an acceptable method to meet L&C#1, subject to a successful outcome of the recombiner testing program to limit the amount of radiolytic gases in support of the final design.

L&C #2—Isolation Condenser Return Line Design and Further Demonstration of Transient Reactor Analysis Code General Electric Modeling Capability

The use of this CEM is limited to a BWRX-300 design that a proper isolation condenser return line layout is chosen, such as a loop seal or a water trap, to prevent reverse flow from RPV into the IC return line throughout a 72-hour period following the event or where an applicant or licensee referencing this report demonstrates that the TRACG code is capable of conservatively modeling the overall ICs heat removal capacity when reverse flow occurs in the IC discharge lines.

L&C#2 will be met by adding a loop seal to each ICS condensate return line to prevent reverse flow from the RPV chimney to the ICS through the return lines after the exit of the ICS return lines are uncovered. It is stated in PSAR Section 15.5.4.5 that for L&C#2, "A loop seal has been added to the design to prevent reverse flow from the RPV into the ICs through the condensate return line." The water that collects in the loop seal prevents steam from flowing to the ICS condensers through the ICS drain lines. The limiting case for that is the small liquid pipe break and the ICS heat removal plot (PSAR Figure 15.5-112), where the conservative small liquid break does not show any indication of a perturbation that would result from reverse steam flow in the ICS drain line. The staff finds the addition of a loop seal to the ICS return piping to be an acceptable method to meet L&C#2 subject to confirming the details of the final design and calculations showing that it prevents reverse flow when the ICS drainpipe exits in the chimney region are uncovered.

L&C #3—Demonstration of No Safety-Significant Break Flow Reversal During the First 72-Hours into the Event

The use of this CEM is limited to a BWRX-300 design in which the PCCS is sized sufficiently large such that a reverse flow from containment back to RPV does not occur during the first 72 hours into the event. The applicant or licensee referencing this report needs to demonstrate that no reverse flow could occur, or any reverse flow that occurs under the most bounding flow reversal conditions resulting in the degradation of IC heat transfer is not safety-significant with respect to the acceptance criteria for the BWRX-300 CEM.

PSAR Section 15.5.4.5 states that the fuel and containment acceptance criteria and L&C#3 (no safety-significant break flow reversal during the first 72 hours) are met. It also states that, "Reverse flow from containment to the RPV will not occur as demonstrated in the small break analyses presented in Subsection 15.5.4.5.4 and shown on Figure 15.5-115." PSAR Figure 15.5-115 presents containment and RPV depressurization for the conservative case of small liquid pipe break, under the "no break" or "no back pressure" assumptions. The staff determined that the PSAR does not provide sufficient information to establish whether L&C#3 was appropriately addressed. The staff determined that the presented material did not investigate the physically realistic potential for reverse flow from the containment to RPV in an un-isolated LOCA scenario with bounding PCCS-ICS trains combination for flow reversal. In such a scenario, if ICS depressurizes the RPV faster than PCCS depressurizes containment, reverse flow from containment to the RPV may occur. Non-condensable materials ingested into the RPV from the containment may collect in the ICS and cause degradation of ICS heat transfer, and as a result, both the RPV and the containment could start repressurizing if the flow reversal were to occur, which may violate GDC 38 accident mitigation requirements. Therefore, the staff requested a demonstration of L&C#3 based on coupled TRACG-GOTHIC calculations using the minimum viable number of PCCS trains and maximum number (three) of ICS trains, biased for maximum heat removal for the largest un-isolated break (to maximize the reverse flow potential under maximum pressure differential between the containment and RPV), or provide a justification as to why a bounding L&C#3 analysis would not be needed.

Considering the safety-significant nature of the reverse flow for long-term cooling with respect to several acceptance criteria in SRP Sections 6.2.1.1A and 6.2.2, both liquid and steam SBLOCA analyses were requested from the applicant. To address the staff's question about the bounding L&C#3 scenario for flow reversal from the containment to RPV in meeting the various acceptance criteria driven by GDCs 16, 50, and 38, and regulatory requirements in SRP Sections 6.2.1.1A and 6.2.1.3, for 72 hours post-accident, the applicant made the GVH report, NEDC-34349P, Revision 0, available for audit in the electronic reading room. The document presented the results of uncoupled TRACG-GOTHIC calculations for limiting steam break SBLOCA with two ICS trains and demonstrated that L&C#3 was met, as there was no flow reversal from the RPV to the containment. The staff conducted an audit of the report and had a clarification call with GVH and TVA. The staff also performed confirmatory analyses of the containment thermal-hydraulic response for 72 hours using coupled/uncoupled TRACE models developed during review of LTR NEDC-33922P-A, "BWRX-300 Containment Evaluation Method." The staff performed and compared both uncoupled (GVH-PSAR method) and coupled (TRACE-Staff concerns) calculations for both small liquid and steam breaks. This showed that in the uncoupled calculations, the RPV pressure drops below the containment pressure in the long term to allow for the flow reversal that is conservative, while in the more realistic coupled calculations, the RPV pressure stays above the containment pressure, thus precluding the flow reversal potential for small breaks. The coupled calculations performed by the NRC staff give sufficient confidence that the information documented in the GVH report, NEDC-34349P, can be supplied in the FSAR and can support meeting L&C#3 for SBLOCA for the final design.

Based on the above, the staff concludes that the information reviewed in the PSAR and during the audit supports issuance of a CP. Staff will confirm these conclusions in the FSAR for the final design. The FSAR should provide justification for break sizes assumed to meet L&C#3 for the most bounding flow reversal conditions and identify the methods that will be used to analyze these breaks. The staff expects the applicant to demonstrate the most limiting number of ICS trains needed for the bounding PCCS-ICS trains combination to meet L&C#3, at the OL stage for the FSAR reactor design.

L&C #4—Demonstration of the Applicability of the BWRX-300 Containment Evaluation Method to the Final Passive Containment Cooling System Licensing-Basis Analysis

The use of this CEM was demonstrated for a BWRX-300 design with the specific concentric-pipe PCCS design configuration and placement described in this LTR. For any alternate PCCS design configuration and placement, the applicability of this method and the PCCS modeling approach must be reviewed and found to be acceptable by the NRC for BWRX-300 licensing-basis analyses.

Figure 6.2-1 in the PSAR presents a new PCCS design that consists of three independent trains each with a passive containment cooling pipe array of condensing pipes, vertically connected to an intake header at the bottom and a return header at the top, for the cooling pools subcooled water circulation. The new PCCS design and piping structure are different from the earlier concentric-pipe PCCS design that was based on individual containment penetrations used in NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," to demonstrate the BWRX-300 CEM. Therefore, as required by L&C#4 in NEDC-33922P-A, the applicability of the BWRX-300 CEM and the modeling approach must be reviewed by the staff and found to be acceptable for the new PCCS design. Though conceptually similar, there are differences between the two

PCCS designs. The earlier PCCS design in the LTR used 10 vertical concentric-pipe PCCS units with annular return section between the inner and outer tubes, while the new PCCS design in the PSAR uses three radiator-style PCCS arrays of eight once-through pipes, with two of the three arrays credited in the PSAR containment safety analyses. A staff audit of the engineering calculations provided some input parameters of the GOTHIC model for BWRX-300 containment with the new PCCS design. The staff also conducted an audit of the GVH report, NEDC-34349P, Revision 0, made available for audit by the applicant in the electronic reading room.

The staff expected the most important phenomenological difference between the two PCCS designs to be in the **[[** **]]** that is rated **[[** **]]** in the PCCS Phenomenon Identification and Ranking Table (PIRT; NEDC-34349P, Table 2.4-2). In comparing the LTR and PSAR PCCS heat exchanger designs, the staff expected the new PSAR design to be less thermally efficient because the older LTR PCCS design would have lower tube-side thermal resistance and higher water heat transfer coefficient caused by the annular tube-side flow. None of the PSAR Chapters 6 and 15 contents, nor the audited material, provided specific results for the new PCCS design to compare its performance with the earlier concentric-pipe PCCS design described in the LTR. However, in reviewing the results of the DBA LOCA containment response documented in PSAR Section 15.5.4.5, the staff noted that the PCCS design described in the PSAR appeared to be more thermally efficient, resulting in lower peak containment pressures and temperatures than compared to similar results presented in the LTR. Based on information provided in the PSAR, the staff could not determine why the new PCCS design appeared to be more thermally efficient, when the earlier PCCS design described in the LTR was expected to be more efficient than the new design.

Considering the safety-significant role of the PCCS in mitigating the long-term containment pressure during an un-isolated SBLOCA, as well as the potential for reverse flow from the containment back into the RPV, the NRC staff requested the applicant justify the PCCS tube-side heat transfer performance for the new PCCS design. The staff suggested that the applicant may consider providing results to compare the two PCCS designs for the PCCS tube-side mass flow rate, heat transfer coefficient, exit temperature, pool temperature, and overall PCCS heat transfer rate, to understand the differences in the PCCS performance and modeling with respect to the CEM applicability. As required by NEDC-33922P-A, the information requested was to assess whether Limitation and Condition #4 has been appropriately addressed for the new PCCS design referenced in the PSAR, and to evaluate the new PCCS design for several containment safety findings for the acceptance criteria in SRP Sections 6.2.1.1A and 6.2.2 for PSAR Sections 6.2.2 and 15.5.4.5. The staff audited supplemental information about the differences between new PCCS design referenced in the PSAR and the earlier concentric-pipe PCCS design that was based on individual containment penetrations used in NEDC-33922P-A to demonstrate the BWRX-300 CEM. During the audit, the staff was able to confirm that the reason for the lower containment pressure in the PSAR resulting from the un-isolated small break cases, as compared to the ones reported in NEDC-33922P-A, Revision 3, is the smaller size of the break used and not a significant increase in heat removal rate modeled for the new PCCS design. The staff accepts this as a reasonable explanation of the reduced containment pressure that is not indicative of modeling more efficient heat transfer in the new PCCS with respect to L&C#4.

The staff also audited a comparison of various heat transfer characteristics of the new PSAR and earlier NEDC-33922P-A concentric-pipe PCCS designs to demonstrate the differences in their thermal performance and modeling approach with respect to the BWRX-300 CEM applicability. During audit, the staff was able to confirm that for comparable containment thermal-hydraulic conditions, the new PCCS design was less efficient, which led to a higher containment pressure. This is conservative and consistent with the reduced role of the forced convection contribution in the new PCCS design as confirmed by the Richardson and Nusselt numbers that show that the convection inside the PCCS tubes is clearly in the natural convection regime. This is another qualitative confirmation by the CEM that the new PCCS design would be less efficient than the earlier concentric-pipe PCCS design due its higher tube-side thermal resistance and lower water heat transfer coefficient. This shows that the most important phenomenological difference between the two PCCS designs (i.e., **[[** **]]** that is rated **[[** **]]** in the PCCS PIRT Table [NEDC-34349P, Table 2.4-2]) is being appropriately captured in the PSAR.

Given the preliminary nature of the design and the scope of the CP review, the staff is unable to make a final regulatory finding on L&C #4. However, for the purpose of issuing a CP, the qualitative trends predicted by the CEM for the new PCCS design are in line with the staff's engineering judgement. Therefore, the staff concludes that the information provided by the applicant is adequate to resolve the PCCS design modification in the PSAR for granting a CP. Considering the safety-significant role of the PCCS in mitigating the long-term containment pressure during an un-isolated SBLOCA as well as the potential for reverse flow from the containment back into the RPV, the conclusion and conformance with the above L&C's will need to be confirmed for the final design for the OL. In addition, the staff will review DEC and PRA analyses in the OL phase to verify that these analyses appropriately address reverse flow impacts on ICS performance following the spectrum of loss of coolant accidents.

15.5.4.3.6 Decrease in Reactor Coolant Inventory (DBA) Event Category (LOCA)- ECCS Performance

<u>Design-Basis LOCAs</u>

The applicant specified that ECCS performance for the DBA LOCA event category also is covered by PSAR Section 15.5.4.5. Such analysis is required by 10 CFR 50.34(a)(4), which specifies 10 CFR 50.46 conformance at the CP stage. As required by 10 CFR 50.46(a)(1)(i), ECCS cooling performance must be calculated in accordance with an acceptable evaluation model. Use of an acceptable evaluation model provides assurance that calculated performance for postulated LOCAs provides a high level of probability that acceptance criteria are not exceeded. An acceptable analysis approach is outlined in NUREG-0800, Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary" (Revision 3). PSAR Table 1.9-15 indicates that PSAR Section 15.5.4.5 conforms with SRP Section 15.6.5. In PSAR Section 15.5.4.5, the applicant stated that the evaluation model described in NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," was used to perform the LOCA DBA safety analysis. The staff notes that the topical report describes an approved method to evaluate the BWRX-300 containment performance and not the ECCS system focused on fuel cooling performance. An analysis of ECCS performance for fuel cooling after a LOCA is contained in the response to RAI 15.5-2 (ML26111A161, non-proprietary; ML26111A162, proprietary), which requested preliminary information in the forms of methodology and analyses regarding fuel-cooling-focused/ECCS-performance LOCA ("ECCS/LOCA") analyses to support issuance of a CP, in the absence of a dedicated, NRC-approved ECCS/LOCA topical report evaluation method.

NUREG-0800, SRP Section 15.6.5 acceptance criterion II.1 states that an evaluation of ECCS performance uses an evaluation model satisfying the requirements of 10 CFR 50.46, RG 1.157 (NRC 1989-TN13259), and Section I of Appendix K to 10 CFR Part 50. Appendix K also specifies documentation required for evaluation models. Acceptance criterion II.1 in SRP Section 15.6.5 is not satisfied by the implementation of NEDC-33922P-A Revision 3 (a) “Containment/LOCA” EM) in PSAR Section 15.5.4.5, despite this being an implementation of something closely related to an ECCS/LOCA EM. One reason for this is that Containment/LOCA LTR prescribes simulation conditions that maximize the challenge to the containment integrity rather than maximizing the challenge to fuel cooling. Further, because NEDC-33922P-A does not conform to the LOCA evaluation model requirements in either 10 CFR 50.46(a)(1)(i) or (ii), it is not an acceptable approach to assessing the LOCA acceptance criteria required by 10 CFR 50.46(b). On its own, NEDC-33922P-A does not cover the correct “uncertainties in the analysis method and inputs [that] must be identified and assessed” (e.g., 10 CFR 50.46(a)(1)(i)) for post-LOCA fuel-cooling-performance). NEDC-33922P-A also is not an implementation of the “required and acceptable features of appendix K ECCS Evaluation Models” (as allowed by 10 CFR 50.46(a)(1)(ii)).

While an approved and applicable ECCS/LOCA EM has not been used for the PSAR, the response to RAI-15.5-2 and preliminary technical and design information included in the PSAR provides sufficient information on the proposed design to support issuing a CP. The staff concluded that further technical and design information needed to complete the safety analysis can be reasonably left for later consideration in the FSAR in accordance with 10 CFR 50.35. Aspects of the BWRX-300 design that support the sufficiency of the preliminary safety analysis results presented in the PSAR include the following:

- reliance on passive safety features such as ICS
- use of inherent margins, such as the large water inventory in the reactor vessel and the ICS pools, to reduce the severity of system challenges
- reduction in the number and size of RPV nozzles
- placement of the RPV nozzles well above the top of active fuel on the RPV
- inclusion of reactor isolation valves (RIVs) attached directly to the RPV (which minimizes the potential for breaks between the reactor vessel and RIVs) and their ability to quickly isolate a break in pipe

An ECCS/LOCA EM acceptable to the NRC will need to be used to demonstrate compliance with 10 CFR 50.46 at the OL stage. The response to RAI-15.5-2 states GVH is targeting a submittal date for the ECCS/LOCA LTR in 2027. The staff considered the impact of the lack of an approved ECCS/LOCA EM at the CP licensing phase. Based on the design factors noted above and the preliminary LOCA evaluations provided in the PSAR and other docketed information, the staff determined that sufficient information was provided by the applicant to adequately describe facility design features associated with ECCS cooling performance. Since an approved ECCS evaluation model that is compliant with 10 CFR 50.46 was not provided as part of the PSAR, an exemption from the requirement under 10 CFR 50.34(a)(4) to provide an analysis and evaluation of ECCS cooling performance in accordance with 10 CFR 50.46, is necessary and discussed later in this section.

As further discussed below, RAI 15.5-2 elicited information that would be adequate to meet 10 CFR 50.46(b) acceptance criteria in a preliminary fashion to support timely issuance of a CP for CRN-1. The RAI requested a description, results, and summary of results from safety analysis simulations where plant parameters[5] and initial conditions[6] are biased to conservatively model initial RPV coolant inventory and the fraction of inventory that is retained in the RPV. Model uncertainties and PIRT multipliers[7] were requested to be set to the least favorable distribution and direction, respectively, from the perspective of post-LOCA fuel cooling. Among other topics, the RAI requested re-running the large FW line break (PSAR Section 15.5.4.5.2), re-running two un-isolated small breaks from PSAR Section 15.5.4.5.4, and running a new small break on an IC condensate line loop seal. Additionally, information regarding breaks on CUW lines was requested.

The RAI response states that a not-yet-specified break detection system would be used in the ICS lines to detect breaks in the vulnerable range and isolate the affected ICS line. The RAI response states that analyses of breaks on ICS return line loop seals and SDC PIPE components, which patch to an IC loop, will be provided in the FSAR. GVH explains that a CUW line break will not be limiting for breaks smaller than the 19 mm break detection threshold because the CUW isolation valves will close on the DL3-36 low RPV water level L2 isolation signal.

According to the RAI response, the limiting break location, in terms of minimum coolant inventory at the PSAR age of design maturity, is the small liquid instrument line break concurrent with a LOPP. The corresponding Containment/LOCA response for this event is described in PSAR Section 15.5.4.5.4, “Small Steam and Liquid Pipe Breaks Inside Containment.” Section 15.5.4.5.4 of the PSAR states “[a]ll liquid pipe break nozzles are at least 4 meters above top of active fuel (TAF). A small pipe break on instrument lines may remain un-isolated indefinitely.” In the RAI response, Table 15.5-2-6 “Summary of Results for Un-isolated Small Liquid Pipe Breaks Concurrent with LOPP, the downcomer collapsed liquid level (CLL) 72 hours after the break is **[[ ]]** from the bottom of surface of the RPV interior volume for the case that ran from an initial downcomer water level of 12 in below the nominal value, L5, of 21.097 m. In the PSAR, downcomer CLL is noted as “reactor water level” or “reactor level.” Although the downcomer CLL is below the TAF (8.877 m above the bottom surface of the RPV interior), RAI response Figure 15.5-2-18 “Peak Cladding Temperature and Saturation Temperature, Unisolated Small Liquid Pipe Break with LOPP” shows the time dependent fuel peak cladding temperature (PCT) never exceeds the initial value; this satisfies the second LOCA acceptance criterion of “fuel cladding temperature will not exceed the normal operating temperature,” which applies when the first criterion of “reactor level will not decrease below TAF” is violated. The RAI response describes “normal operating temperature range” as variable according to plant conditions, with different ranges for different conditions (e.g., power level). According to the response, this second-tier BWRX-300 LOCA acceptance criterion on fuel cladding temperature is satisfied when the fuel cladding temperature remains less than the initial steady state cladding temperature calculated at 100 percent power for a given event.

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[5] In this case, plant parameters include things such as protection system setpoints, valve capacities, etc., that influence the characteristics of the transient response but do not (when properly prescribed) have an impact on steady-state operation (GVH’s terminology).
[6] In this case, an initial condition is key plant input that determine the overall steady-state nuclear and hydraulic conditions prior to the transient. This input is essential to determining that a steady-state plant condition has been established (GVH’s terminology).
[7] In this case, biases and uncertainties are indicated by data comparisons between separate effects test data and TRACG calculations performed with the best-estimate version of the code that are used to establish PDFs for TRACG parameters and correlations. Biases are compensated by appropriate choice of the mean value of the PIRT multiplier and uncertainties are accommodated by choosing PDFs to represent the standard deviation of the data comparisons (GVH’s terminology).

The fuel still has cooling despite the downcomer CLL being lower than TAF because the water inside the fuel bundles is in a two-phase state (liquid and vapor) and the level swell keeps the fuel covered with a two-phase mixture. While an in-shroud CLL could be computed, it would mask the fact that the two-phase nature of the in-shroud water causes liquid water to be present at higher elevations than if there were a sharp liquid-vapor interface in the coolant inside the shroud. RAI response Figure 15.5-2-25 “Void Fraction in the Top Node of the Heated Section of the Fuel Channels, Small Liquid Pipe Break, Initial Level: 12 in. Below Nominal,” shows that the channel boxes have at least **[[ ]]** liquid by volume for most of the 72 hours after the LOCA initiation, after an initial decease in the void percentage after the LOCA. While the void percentage is **[[ ]]**, the flow speed in the core post-scram and post-ICS actuation is very low and there is very little in-RPV recirculation (i.e., relatively weak mass flow into the bottom of the fuel channels). This is a confined pool boiling heat transfer mode as opposed to the confined flow boiling heat transfer mode that exists during normal BWR operation. Pool boiling is a favorable heat transfer mode in this context, and ICS condensate return to the chimney rather than the downcomer reinforces the pool boiling aspect for BWRX ECCS core cooling. The code predicts sufficient liquid is in contact with the decay-power-heated fuel rod cladding to allow for heat removal from the cladding via coolant vaporization.

The RAI response clarifies that when the PSAR uses the phrase “fuel overheat,” the intent is not to define a specific threshold for “fuel overheat” but to emphasize that the BWRX-300 LOCA acceptance criteria are conservative and that consideration of uncertainties for each 10 CFR 50.46(b) criterion is not needed. With respect to the degree of conservatism in these acceptance criteria as it would relate to rendering uncertainty analysis unnecessary, the staff makes no finding at this time but will address this issue during the review of the formal BWRX-300 ECCS/LOCA LTR. The generalities of the protype ECCS/LOCA EM, including to the importances of PIRT multipliers (see NEDC-34043P, Revision 1 and the review of this NEDC in Section 15.11 of this report), can be found in the PSAR under section 15.5.1.2.1 “TRACG.” The RAI response presented a sample sensitivity analysis regarding the influence of interfacial shear in the core region. As the RAI response explains the details of this sensitivity study on interfacial shear in the core, the staff concluded that that the associated PIRT parameter has a modest influence on the void fraction in the uppermost heated cell in the hot channel (RAI response Figure 15.5-2-27). The formal BWRX-300 ECCS/LOCA LTR is expected to quantify uncertainty on identified high importance PIRT multipliers.

The RAI response provides TRACG code qualification in support of ECCS/LOCA analyses for BWRX-300 beyond what is found or referenced in NEDC-34043P, Revision 1. TRACG has been qualified for predecessor GVH designs such as ESBWR and SBWR. Because of differences between the BWRX-300 design and predecessor designs, prior TRACG qualification to the ESBWR and SBWR is not fully applicable to the BWRX-300. For example, the SBWR and Certified Design ESBWR had partitions in the chimney, but BWRX-300 does not. In addition, ESBWR ICs returned condensate to the downcomer while BWRX-300 ICs return condensate to the chimney. Hence, the expanded information regarding TRACG code qualification for BWRX-300. A qualification program of note in the RAI response is benchmarking work between computational fluid dynamics, TRACG, and a scaled experiment of the ICS condensate return to the chimney. It is expected that this additional code qualification information will be part of the formal ECCS/LOCA LTR.

On the break sensitivity in the RAI response, staff notes that there are some unexpected trends on liquid-space break in RAI Response Figure 15.5-2-25, which will be revisited during the review of the formal ECCS/LOCA EM and/or the FSAR. Specifically, the staff will need to understand why CCFL breakdown and flooding occurred in the CHAN components in **[[**

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components based on experimental data from the SSTF test facility and LOCA calculations in previously licensed reactors. This will be resolved during the review of the ECCS/LOCA EM required for the OLA.

In the RAI response, the staff also noted GVH's discussion regarding whether LOPP-LOCA (a LOCA during which non-safety-related preferred power is unavailable) or non-LOPP-LOCA (during which preferred power remains available) would be more limiting in terms of retained water mass in the RPV. GVH stated that in a non-LOPP-LOCA, the CRD purge flow and reactor level control (RLC) would continue to operate. The water mass ejection into containment would be prolonged until such time as the containment inerting system (CIS) and PCCS become insufficient to prevent containment pressurization. An eventual high containment pressure signal would cause a hydraulic or diverse hydraulic scram, ICS actuation, RPV and containment isolation (see PSAR tables 7.3-1 "Safety Category 1 Instrumentation and Control Functions and Initiation," 7.3-3 "Safety Category 2 Diverse Protection System Functions and Initiating Signals," or 15.5-44 "DL3 Functions Credited in Conservative LOCA Analyses")[8]. It is possible for a small break mass and energy release remains within the capabilities of the CIS and PCCS, thus never meeting the high containment pressure signal. At present, the PSAR states "[f]uel heat up does not occur even without injection to the RPV" in section 15.5.4.5.4, "Small Steam and Liquid Pipe Breaks Inside Containment," which is a DBA subsection. PSAR DEC subsection 15.6.3.12, "Systems Credited in the Level 1 Probabilistic Safety Assessment," specifies the CRD Hydraulic Subsystem can provide high pressure injection to the reactor. The RAI response presents this injection function as a mitigating system for non-LOPP-LOCAs. It would not be permissible in a Chapter 15 DBA analysis to credit mitigation functions provided by a function that is not safety-related unless it made the scenario more limiting than if the function was not credited. Because Chapter 15 DBE safety analyses mandate crediting only safety-related SSCs, during the review of the formal ECCS/LOCA EM, for completeness, staff will ensure the reactor system response to a non-LOPP LOCA that only features safety-related SSC effects.

The applicant's response to RAI-15.5-2 provides a sufficient preliminary analysis at the CP stage showing that the system design provides adequate fuel cooling ECCS/LOCA performance because the results in the RAI response indicate significant margin to the LOCA acceptance criteria on reactor water level and/or fuel cladding temperature.

PSAR Table 15.5-44 describes several "line break indication" functions intended to detect breaks in the MS, FW, SDC, CUW, and ICS systems. These safety-related functions rapidly isolate the RPV from a detected line break. During the regulatory audit, the applicant explained that information about how the "line break indication" system works will not be ready for review until the OLA. The staff acknowledges that the specific details of how line breaks would be detected for each system are not required for the PSAR. However, the LOCA safety analysis does not simulate DL3 line break indication functions but instead assumes that line break isolation begins within either 1 or 5 seconds (see PSAR Tables 15.5-22 through 15.5.25). The 1 second and 5 second timing requirements should be considered as specifications that need to be met for line break indication functions and verified at the OL stage to see if the final design

[8] Note an overlap between PSAR tables 7.3-1 (SC1) and 7.3-3 (SC2). Table 7.3-1 has a scram for "LOW RPV water level (L3)" and Table 7.3-3 has a scram for "LOW reactor water level (L3). This overlap is related to Table 7.3-3 being denoted as "diverse protection systems (DPS)," indicating there is more than one scram signal (i.e., an SC1 and an SC2) on reactor water level L3.

numbers based on the actual implementation bound those numbers in a conservative way. The staff observed that PSAR Figure 15.5-83 shows reverse flow in the ICS system. Multiple DBA events analyzed in the PSAR fall into the category of having significant reverse flow in the ICS return line that could lead to ICS isolation. For example, the loss of preferred power DBA has significant reverse flow in the ICS return line. The PSAR and associated docketed information does not provide a detailed description of how leakage will be detected in an ICS train and the control logic that will be used to initiate isolation of an ICS train when excessive leakage is detected. Therefore, the staff was unable to evaluate if potential ICS flow reversals could lead to inadvertent isolation of an ICS train. The staff will evaluate the "line break indication" system and how it interacts with the ICS when it is eventually described; this will be during the OLA/FSAR review and possibly also during the review of the formal ECCS/LOCA LTR. Breaks in the SDC and boron injection system lines could also challenge RPV inventory because these lines connect to the ICS drain lines; breaks in these systems will be evaluated during review of the OLA/FSAR.

As noted above, the staff has concluded that the preliminary DBA LOCA analysis presented in the PSAR is sufficient to describe major features and principal architectural and engineering criteria for the design. Furthermore, the staff has determined that technical and design information, including development of an ECCS evaluation model meeting the requirements of 10 CFR 50.46, can be reasonably left for later consideration at the OL licensing phase and will be reviewed with the FSAR, in accordance with 10 CFR 50.35. Since an approved ECCS evaluation model that is compliant with 10 CFR 50.46 was not provided as part of the PSAR, an exemption from the requirement under 10 CFR 50.34(a)(4) to provide an analysis and evaluation of ECCS cooling performance in accordance with 10 CFR 50.46, is necessary as further discussed below.

Exemption from 10 CFR 50.34(a)(4) and 50.46(a)(1)

10 CFR 50.34(a)(4) requires that a construction permit application include a PSAR that includes an analysis and evaluation of ECCS cooling performance performed in accordance with the requirements of 10 CFR 50.46. 10 CFR 50.46(a)(1) requires that ECCS cooling performance be calculated in accordance with an acceptable evaluation model. The evaluation model must meet one of two alternative requirements. Under 10 CFR 50.46(a)(1)(i), the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in 10 CFR 50.46(b), there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II, "Required Documentation," sets forth the documentation requirements for each evaluation model. Alternatively, under 10 CFR 50.46(a)(1)(ii), an ECCS evaluation model may be developed in conformance with the required and acceptable features of 10 CFR 50, Appendix K, "ECCS Evaluation Models."

As noted above, since the applicant did not submit an acceptable ECCS evaluation model in the PSAR, the staff is initiating an exemption for the Clinch River CPA from the requirements of 10 CFR 50.34(a)(4) and 10 CFR 50.46(a)(1) that the PSAR contain analysis and evaluation of ECCS cooling performance where ECCS cooling performance is calculated in accordance with an acceptable evaluation model.

Pursuant to 10 CFR 50.12, "Specific exemptions," the Commission may upon its own initiative grant an exemption from the requirements of the regulations of 10 CFR Part 50 when: (1) the exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request.

Authorized by Law

As stated above, 10 CFR Part 50 allows the NRC to grant exemptions. The NRC staff has determined that the proposed exemption will not result in a violation of the Atomic Energy Act (AEA) of 1954, as amended, or any of the Commission's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the staff has determined that the proposed exemption is authorized by law.

No Undue Risk to Public Health and Safety

The staff considered the LOCA and ECCS cooling performance analysis and evaluations presented in the PSAR and other docketed information supporting the CP application. The applicant has provided preliminary evaluations and assessments of LOCA design basis events using methods and tools that have been generally applied to similar evaluations. The LOCA mitigation strategy used for the proposed design is unique in that it relies on rapid isolation of postulated reactor coolant system breaks to preserve cooling water inventory in the reactor core and passive cooling systems to remove decay heat. Therefore, the proposed design does not rely on the active front line safety systems and associated electrical and mechanical support systems typically used in previously licensed light-water reactors to provide post-LOCA makeup water and decay heat removal. Based on a review of these preliminary evaluations and assessments, the staff determined that the LOCA conditions analyzed in the PSAR are sufficient to identify major features and principal architectural and design criteria related to ECCS cooling performance, including safety functions associated with ICS cooling performance and reactor vessel isolation capability. Therefore, the staff has determined that the applicant has provided sufficient information to ensure that the ECCS features for the proposed facility can be constructed at the proposed location without undue risk to public health and safety.

Further, the construction permit, if issued, will not allow possession or use of special nuclear material or facility operation, and therefore it is not possible for the lack of an acceptable ECCS evaluation model to result in a failure to adequately cool fuel following a LOCA. Before the applicant may possess special nuclear material, the applicant will have to apply for and receive a license under 10 CFR Part 70. Further, before the applicant is able to operate the proposed facility it will have to apply for and receive an operating license under 10 CFR Part 50. As part of receiving an operating license, the applicant will have to submit a final safety analysis report which, under 10 CFR 50.34(b)(4), shall include analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents performed in accordance with the requirements of 10 CFR 50.46. Therefore, the applicant will eventually have to provide this information. Consequently, the staff determined that not providing this information at this time does not pose an undue risk to public health and safety.

Consistent with Common Defense and Security

Under 10 CFR 50.34, CP applicants do not need to provide a physical security plan as part of their applications. Thus, as required by 10 CFR 50.12(a)(1), the staff has determined that granting this exemption is consistent with the common defense and security.

Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), exist because the application of the regulations in 10 CFR 50.34(a)(4) and 10 CFR 50.46(a)(1) are not necessary to achieve the underlying purpose of the rule. For a CP applicant, the primary objective of analyzing and evaluating ECCS cooling performance using an acceptable evaluation model is to ensure that the design incorporates an ECCS system of sufficient capability to provide abundant core cooling. For previously licensed plants, the ECCS system typically includes multiple trains of active cooling water makeup systems, onsite electrical distribution, and support systems to provide essential cooling water. The design of such systems requires use of an acceptable evaluation model to ensure that active system capabilities are appropriately sized and designed (e.g., flow rates, electrical load sequencing, and heat removal capabilities) to provide adequate margin to fuel acceptance criteria.

The proposed design uses a different LOCA mitigation strategy that focuses on rapid break isolation and use of a passive cooling system with substantial heat removal capabilities. The staff notes that the applicant evaluated LOCAs and ECCS performance using recognized codes that have previously been used in acceptable evaluation models and that these codes are capable of calculating the phenomena that are important to ECCS calculations for the proposed design. Therefore, because of this simpler approach and based on the preliminary analysis provided in the PSAR, the staff determined that the proposed ECCS systems can adequately be sized and designed to support construction activities without an evaluation model fully meeting the requirements of 10 CFR 50.46(a)(1).

Further, the CP will not authorize possession or use of fuel, so there is no potential for a LOCA resulting in fuel exceeding the 10 CFR 50.46(b) acceptance criteria prior to issuance of an OL. At the OL phase, the applicant will need to meet the requirements of 10 CFR 50.34(b)(4), which will require analysis and evaluation of ECCS cooling performance using an acceptable evaluation model. As noted elsewhere in this report section, the applicant has informed the staff of its planned schedule to develop acceptable ECCS evaluation model to support a future OL application.

The staff also considered the special circumstances described in 10 CFR 50.12(a)(2)(vi) related to other material circumstances not considered when the regulation was adopted for which it would be in the public interest to grant the exemption. As discussed above, the proposed design uses a different LOCA mitigation strategy than designs for which the agency previously issued a CP. More specifically, the proposed design focuses on rapid break isolation and use of a passive cooling system with substantial heat removal capabilities. The requirements in 10 CFR 50.34(a)(4) and 50.46(a)(1) pre-date this type of design, meaning the NRC did not consider a design with those features when adopting these requirements. While use of an acceptable ECCS evaluation model can be useful for some designs when reviewing LOCA mitigation capability at the CP stage, for small modular reactor ECCS systems with reduced reliance on active systems, there are alternate methods to identify the major features and principal architectural and engineering criteria for the design. In this instance the applicant provided preliminary evaluations and assessments that allowed the NRC staff to identify major features and principal architectural and design criteria related to ECCS cooling performance, including safety functions associated with ICS cooling performance and reactor vessel isolation capability, to support issuance of the CP. Therefore, in light of the material circumstances not present when the Commission adopted the relevant requirements, and considering the evaluations and assessments the applicant submitted, the staff determined that granting an exemption for the CP review would be in the public interest because it would promote efficient use of limited government resources without causing a commensurate negative effect on the radiological safety. As noted above, the staff will review ECCS cooling performance, calculated with an acceptable evaluation model, during the OL review.

Therefore, the staff concludes that the special circumstance prescribed in 10 CFR 50.12(a)(2)(ii) and 10 CFR 50.12(a)(2)(vi) are present.

Environmental Considerations

The staff evaluated the impact of granting this exception on the environmental review conducted for the CP application. The Supplement Environmental Impact Statement (SEIS) (NRC 2026-TN13055) addresses the environmental impacts from postulated accidents, including LOCAs. The staff determined that the exemption would not change the assumptions or conclusions documented in the SEIS.

Excluding Certain LOCA Beak Locations from the Design Basis

Appendix A, GDC 35, of 10 CFR Part 50, requires a system to provide abundant emergency core cooling by transferring heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Appendix A of 10 CFR Part 50, defines loss of coolant accidents as, “those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.” Regulations in 10 CFR 50.46 implement GDC 35 and require, in part, that the ECCS’s cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, location, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Finally, 10 CFR 50.34(a)(4) requires CP applications to include an analysis and evaluation of ECCS cooling performance following postulated LOCAs in accordance with the requirements of 10 CFR 50.46. During its review of the PSAR, the staff observed that the bounding LOCA events analyzed identified in Table 15.2-2, and as evaluated in subsection 15.5.4.5, do not consider the loss of coolant from the bolted flanged connection between the reactor vessel and reactor isolation valves. A loss of coolant at one of these locations essentially results in an un-isolated large break LOCA scenario that can significantly challenge core cooling.

The NRC staff raised this concern during an earlier review of NEDC-33934P. In response, as documented in NEDC-33934P, Revision 2, Section 7.4, GVH stated that considering a break from the connection between the RPV assembly and RPV isolation valve assemblies is inconsistent with NRC approval of NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel (RPV) Isolation and Overpressure Protection Licensing Topical Report.” The staff clarifies that the NRC’s approval of the material in NEDC-33910P-A related to 10 CFR 50.46 is with respect to 10 CFR 50.46(b) only and the proposal to use more restrictive surrogate acceptance criteria in lieu of the 10 CFR 50.46(b) ECCS cooling performance criteria. The NRC staff provided no approval in NEDC-33910P-A related to any other LOCA requirements, including requirements of an acceptable evaluation model for the calculation of the most severe LOCA in accordance with

10 CFR 50.46(a)(1). The staff further concluded that if an applicant for a CP under 10 CFR Part 50, or a DC or COL under 10 CFR Part 52 (TN251), is not able to demonstrate compliance with an NRC regulation when the detailed design of the BWRX-300 SMR is complete, the applicant will be expected to justify an exemption from the applicable regulatory requirement. The NRC staff will evaluate the regulatory compliance of the final design of the RPV isolation and overpressure protection features for the BWRX-300 SMR during future licensing activities in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable.

The response in NEDC-33934P, Section 7.4, also states that considering a break from the connection between the RPV assembly and RPV isolation valve assemblies is inconsistent with plain language and regulatory history because 10 CFR 50.46 is only subject to "breaks in pipes." The staff disagrees with this perspective and clarifies that the underlying purpose of 10 CFR 50.46, which includes prescriptive criteria that support implementation of GDC 35, is to ensure the ECCS is appropriately sized such that the core will remain sufficiently cooled following the arbitrary or hypothetical loss of coolant from the RCPB. 10 CFR 50.46(a)(1)(i) pertains to the calculational framework and spectrum of LOCAs to be analyzed, which intends to ensure that the most severe LOCAs are calculated and that uncertainty is accounted for such that there is a high probability abundant core cooling will be provided. During promulgation of 10 CFR 50.46, the Commission ensured a strong balance between prevention of accidents and mitigation capabilities using implicit probabilistic considerations. The Commission acknowledged that traditional RPVs are sufficiently robust so the ECCS need not be sized for catastrophic failure or rupture of the vessel. Likewise, it is understood that only those locations that result in the most severe consequences need to be analyzed and not every location or size must be explicitly analyzed. Codification to define LOCAs as "breaks in pipes" within 10 CFR 50.46 was based on the longstanding practice of traditional large LWR designs to size the ECCSs due to failures in RCS piping. This yields a simplified bounding approach that also addresses other non-limiting component failures and break sizes in the RCPB. For example, designing an ECCS to an arbitrary loss of coolant from the largest pipe in the RCS inherently covered uncertainties and unknown failure mechanisms for components or connections (e.g., component manways, valve bodies) eliminating the need to explicitly analyze them. This robust approach to LOCA evaluations ensured mitigation capabilities exist for a sufficiently broad spectrum of failures within the RCPB. In combination with means of prevention via application of very high quality standards, defense-in-depth is maintained, and reasonable assurance of adequate protection of public health and safety is provided. Based on the historical interpretation and implementation of 10 CFR 50.46, and other recent SMR reviews, the staff determined that these components and connections within the RCPB require a LOCA evaluation consistent with the aforementioned LOCA-related regulatory requirements.

Notwithstanding the above, risk-informed alternatives to traditional compliance with 10 CFR 50.46 for documenting regulatory exemptions to 10 CFR 50.46(a)(1)(i) and GDC 35 for SMR designs that utilize a similar RPV isolation strategy have been approved by the NRC staff. An applicant choosing to use an alternate approach to compliance with 10 CFR 50.46 should describe the considerations and acceptance criteria for determining whether plant design features, materials, and operational controls provide sufficient confidence that such break locations are highly unlikely and therefore need not be treated as design-basis accidents.

In a letter dated March 23, 2026 (NRC 2026-TN13174, non-proprietary; ML26082A196 proprietary), the applicant modified the PSAR to include design characteristics, attributes, and commitments such that breaks in the RPV nozzles, RIV bodies, and associated flanges and connections, are highly unlikely such that postulation of large LOCAs can be excluded from the design bases.

Specifically, the applicant incorporated the following into PSAR Section 5.4.3:

- RIVs and their flanges consist of a single integral forging with no welds and are designed in accordance with ASME Code, Section III, NB-3200, NB-3500, ASME B16.10 and ASME B16.34.
- The flanges of the RIVs, RPV nozzles, and valve-to-pipe transition flange are designed in accordance with ASME Code, Section III, NB-3200, NB-3300, and ASME B16.5.
- Bolting is designed in accordance with ASME Code, Section III, NB-3200.
- The materials of construction for the RIVs and their associated flanges are selected to resist degradation mechanisms and fabricated by forging methods to minimize defects and their propagation. Bolting materials follow the guidance of Regulatory Guide 1.65, Revision 1.
- Stress and fatigue analyses are performed for the RIVs, their associated flanges and bolting in accordance with ASME Code, Section III, NB-3200.
- All applicable loads are considered, and limits established, in accordance with ASME Code, Section III, NCA-2140.
- All RIV bodies are surface and volumetrically inspected in accordance with ASME Code, Section III during fabrication.

The applicant also included the following as provisions for protection and detection of degradation:

- As described in Subsection 5.2.3.2.1, water chemistry control is established with input from Electric Power Research Institute (EPRI) BWRVIP-190, BWR Water Chemistry Guidelines.
- As Described in Subsection 5.2.5, detection of leakage from the RIVs and their associated flanges is provided by multiple methods for monitoring leakage inside containment. The response time of the leakage detection instrumentation inside containment is consistent with RG 1.45 (NRC 2008-TN9209).
- Pre-service and in-service inspections of the RIVs and their associated flanges including all flange bolting are conducted in accordance with the requirements of ASME Code, Section XI. Inboard and outboard flange bolting of RIVs is inspected during each examination cycle and cannot be deferred until valve disassembly or the end of an inspection interval. All RIV flange bolting is volumetrically examined, and no exemptions for size or sampling are used.

The applicant also included a discussion on preliminary PRA:

> A preliminary probabilistic risk assessment of rupture at the RIVs and their associated flanges has been performed and concluded that the frequency of a break at these locations (i.e., the sum of breaks at the flanged RPV/RIV connections and ruptures of the RIV bodies for all large break locations) is less than 1E-05 per reactor-year, with consideration of uncertainty.

The RIVs and their associated flanges are single integral forgings. The staff finds that this demonstrates that the valve bodies have a low probability of failure because no welds are present. The staff also finds that the materials of construction, evaluated in Section 5.2.3 of this report, also provide assurance of low probability of failure of these locations by using forgings and minimizing defects. Pre-service and in-service inspection of all RIV flange bolts, regardless of size or ASME Code exemptions, is acceptable to the staff, because it provides assurance that if degradation exists, it will be detected before propagation of the defects would lead to failure of these locations.

The staff agrees with the applicant's response that RIV bodies are surface and volumetrically inspected in accordance with ASME Code, Section III, during fabrication. However, there are exemptions in ASME Code, Section III, Paragraph NB-2510, that do not require all valve bodies to be volumetrically inspected. Therefore, the augmented volumetric examination should include all valve bodies in the break exclusion zone to provide assurance that defects in the valve bodies during fabrication are minimized so that there is adequate assurance that the probability of rupture is extremely low. Each risk-significant location of interest will have inspections performed by methods that are qualified and can demonstrate that flaws/degradation will be detected and addressed to justify that the proposed design reduces the overall risk of LOCAs through prevention and mitigation. Therefore, a one-time volumetric examination method (radiography or ultrasonic) of all RIV valve bodies prior to startup, regardless of their size due to the risk-significance of these locations, provides justification that these break locations (i.e., RIVs) as beyond-design-basis LOCAs.

Therefore, the staff developed Permit Condition 15-1, which requires the applicant include that the RIV valve bodies, including RIV valves NPS 2, are subject to a one-time augmented volumetric examination during construction (prior to initial startup) using the procedures and acceptance criteria of ASME Code, Section III, Paragraph NB-2540. Certain defects within the valve body volume could grow by mechanisms such as fatigue and compromise the integrity of the valve, which does not provide adequate assurance of low likelihood of failure. Volumetric examination of the valve body volume provides assurance that defects in the valve body volume that could compromise its integrity during its operation are detected and addressed prior to operation.

Permit Condition 15-1 is as follows:

> As a condition of this construction permit, the applicant shall ensure that reactor isolation valve bodies, including reactor isolation valves NPS 2, are subject to a one-time augmented volumetric examination during construction (prior to initial startup) using the procedures and acceptance criteria of ASME Code, Section III, Paragraph NB-2540.

The staff's evaluation of the mechanical framework discussed above, which contributes to the applicant's conclusion of highly unlikely failures from the subject locations, is provided in Section 3.6.1.3 of this report.

Regarding a consequence analysis to demonstrate that dose consequences of a break at the subject location are minimal, the applicant included the following acceptance criteria in PSAR Section 15.2.4.6.4:

- The results demonstrate that the criteria in 10 CFR 50.46(b) are satisfied.
- Offsite dose consequences of the event are demonstrated to meet the following criteria (either by realistic evaluation of the event or showing that the consequences are bounded by another event):
    - An individual located at any point on the boundary of the exclusion are for any 2-hour period following onset of the postulated fission product release would

not receive a radiation dose in excess of 25 Roentgen equivalent man (rem) total effective dose equivalent (TEDE).

- An individual located at any point on the outer boundary of the LPZ, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25 rem TEDE.
- Results demonstrate that the plant safety goals are met.

The PSAR also specifies that the FSAR will perform an EX-DSA for the most limiting un-isolated large pipe break using best estimate or realistic analyses conditions and provides a description of how the analyses will be performed including assumptions for single failures, loss of power, and SSC availability; however, the PSAR does not include the analysis results. Thus, the PSAR is missing the demonstration that the fuel remains within acceptable design limits (i.e., satisfies 10 CFR 50.46(b) performance criteria) for losses of coolant from the connection between the RPV assembly and RPV isolation valve assemblies. Therefore, the staff requested preliminary analysis for these LOCA scenarios of the BWRX-300 design.

In a letter dated March 23, 2026 (NRC 2026-TN13174, non-proprietary; ML26082A196 proprietary), the applicant provided a summary of the preliminary analyses it performed that demonstrates the BWRX-300 design is capable of mitigating the breaks of concern. Specifically, the response includes a description of the analysis of an un-isolated large steam line break concurrent with a LOPP, including sequence of events, initial condition and plant parameter assumptions, identification of mitigating SSCs, and a plot of the water level and fuel temperature results. The response concludes that the event does not challenge the acceptance criteria of 10 CFR 50.46(b), and the radiological consequences are bounded by the postulated maximum hypothetical accident (MHA) presented in PSAR Table 15C-6.

The staff reviewed the applicant’s proposed strategy for excluding certain LOCA break locations from the design basis as an alternative to meeting 10 CFR 50.46 and 10 CFR Part 50, Appendix A, requirements. The staff also evaluated the applicant’s description of its approach to the consequence analysis and corresponding docketed results for losses of coolant from the connection between the RPV assembly and RPV isolation valve assemblies.

Based on the information provided, the staff concludes that the proposed strategy and docketed preliminary results provide sufficient justification that the underlying purpose of the rule could be met. Specifically, an approach that ensures gross ruptures and failures resulting in a LOCA between the RPV assembly and RPV isolation valve assemblies remains highly unlikely through design enhancements while maintaining a LOCA analysis that demonstrates mitigation capability to cool the core, protect containment integrity, and minimize dose consequences, could provide an alternate means to address ECCS and LOCA requirements. This is consistent with past precedent for excluding certain break locations from LOCA consideration (e.g., NRC 2025-TN13173) and the staff concluded that the preliminary information presented at the PSAR stage is sufficient to support issuance of the CP. However, the staff notes that this approach is not consistent with requirements in 10 CFR 50.46 and GDC 35 related to consideration of the most severe loss of coolant accidents. Therefore, at the OL stage, the applicant will need to either demonstrate compliance with 10 CFR 50.46 or seek an exemption in accordance with 10 CFR 50.12, “Specific exemptions,” to adopt the proposed approach. Given the preliminary nature of the safety analysis and design at the CP phase, the NRC staff determined that consideration of an exemption, if necessary, is more appropriate at the OL stage. Therefore, a future FSAR submittal will need to provide the detailed technical bases that either demonstrate compliance with 10 CFR 50.46 or justify an acceptable alternate approach, including sensitivity studies to demonstrate uncertainties and to ensure there are no cliff-edge effects. The staff will review this information at the OL phase and also consider the need for a specific exemption to 10 CFR 50.46 and GDC 35 at that time.

15.5.4.4 Conclusion

The staff concludes that the preliminary BWRX-300 plant design, analysis, and evaluation of DBAs, as described in PSAR Section 15.5.4, with consideration of the exemption related to ECCS/LOCA EM, is consistent with the requirements in 10 CFR 50.34(a), and is sufficient to enable the applicant to describe the principal architectural and engineering criteria for the design and identify major features incorporated for the protection of public health and safety as required by 10 CFR 50.35. Therefore, the NRC staff finds that the information in PSAR Section 15.5.4 adequately supports the issuance of a CP pursuant to 10 CFR 50.35. The staff has recommended approval of an exemption from the requirements of 10 CFR 50.34(a)(4) and 10 CFR 50.46(a)(1), related to the calculation of ECCS cooling performance using an approved evaluation model applicable. The staff has determined that submittal of the ECCS/LOCA EM can reasonably be left for later consideration at the OL licensing phase The staff expects that the ECCS/LOCA EM to be submitted in support of the OL, would be developed consistently with RG 1.203 (NRC 2005-TN13304), or an appropriate alternate approach, and will include a safety analysis code model that is suitable for extended post-LOCA fuel cooling simulations.

For LOCAs designated as beyond-design-basis as discussed above, with consideration for the identified Permit Condition 15-1 associated with reactor isolation valve inspections, the staff finds the information provided by the applicant is consistent with the requirements of 10 CFR 50.34(a) and sufficient for issuance of a CP in accordance with 50.35. An exemption to the requirements of 50.46 and GDC 35 will be considered as part of the OL licensing phase.

15.5.5 Analysis of Design Extension Conditions Without Core Damage

15.5.5.1 Introduction

This section presents the deterministic safety analysis of DECs without core damage for the proposed design as described in the CRN CPA.

The approach for identifying and analyzing DECs is described in PSAR Sections 15.1 through 15.3 and includes identification of PIEs, event categorization, and event grouping based on their frequency and effect on plant parameters.

As described in PSAR Section 15.1.4, the analysis for DEC events without core damage aims to demonstrate that radiological releases associated with reactor coolant remain within acceptable limits and supports the PSA conclusion of no core damage. As described in the PSAR, the primary objectives of the DEC analyses without core damage are to demonstrate (1) the establishment of a second functional DL for DBA PIEs in which DL3 functions were credited in both AOO and DBA analyses and (2) the capability of the plant to avoid core damage under complex or very unlikely event scenarios involving combinations or types of mitigation failures beyond those deterministically postulated.

The PSAR stated that the analysis might credit unaffected functions in DL2, DL3, and DL4a. Unaffected functions are defined as those not impaired by the initiating failure, postulated mitigation failures, or environmental conditions present during the event sequence. The analysis includes DECs without core damage for the following event groups: increase in reactor pressure, reactivity and power distribution anomalies, and decrease in reactor coolant inventory. For the decrease in reactor coolant temperature and increase in reactor coolant inventory event groups, no DECs without core damage were selected, because, as stated, two DLs were already established in the baseline and conservative DSA analyses. The PSAR indicated that further LOCA-related DEC evaluations will be provided in the FSAR.

15.5.5.2 Regulatory Evaluation

The staff based its review on the relevant requirements of the following regulations and guidance:

- 10 CFR 50.34(a)(4) (TN249)
- 10 CFR 50.35

15.5.5.3 Technical Evaluation

Pressure Increase DECs

PSAR Section 15.5.5.2 evaluates pressure increase DECs without core damage assuming failure of the hydraulic scram system, consistent with the Safety Strategy. Under these conditions, reactivity control relies on the CRDM run-in function (DL4a), which provides slower negative reactivity insertion than hydraulic scram and therefore represents a more limiting challenge to fuel cladding and the RCPB. The staff recognizes that this approach is intended to address ATWS-type conditions within the defense-in-depth framework. Consistent with 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” applicants shall submit information in its FSAR demonstrating how it will comply with the regulation. Applicants of a CP are not required to demonstrate compliance with the 10 CFR 50.62 ATWS rule.

PSAR Section 15.5.5.2 identifies several potentially limiting event sequences associated with a failure of hydraulic scram, including closure of one Main Steam Isolation Valve (PSAR Section 15.5.5.2.1), loss of condenser vacuum (PSAR Section 15.5.5.2.3), and loss of preferred power (PSAR Section 15.5.5.2.4). These events represent plausible challenges under hydraulic scram failure conditions for ATWS-type conditions. The staff notes that additional description of the screening methodology and associated acceptance metrics used to identify the bounding DEC without core damage event would further support confirmation that the selected events appropriately bound credible initiating events with hydraulic scram failure.

PSAR Section 15.5.5.2 evaluates a generator load rejection or turbine trip event assuming that one-half of the highest worth control rods fail to scram and that the CRD run-in function fails to insert those rods, with no additional failures postulated. This scenario presents a distinct and potentially limiting reactivity control challenge by assuming loss of both hydraulic scram and backup motor-driven insertion for a substantial fraction of high-worth rods. The staff notes that additional methodological detail, clearly defined quantitative acceptance criteria, and supporting justification describing how this configuration bounds other credible rod failure patterns would further support the conclusion that core damage is prevented for these DEC analyses.

As discussed in Chapter 4, Appendix 4A of this report, the evaluation of thermal-hydraulic instability during ATWS scenarios has not been provided in the PSAR, as the CP applicant is not required to demonstrate compliance with the ATWS rule; therefore, information demonstrating compliance with the ATWS rule is needed at the OL phase. The PSAR currently provides limited description of the methodology used to evaluate ATWS conditions, including the use of TRACG to evaluate expected Type II instability conditions during delayed rod insertion. Additional information describing the evaluation methodology, including justification for the applicability of the analytical tools (such as TRACG) and supporting assumptions related to credited instrumentation functions and actuation timing, would further support the staff's review of these analyses at the OL stage.

Reactivity and Power Distribution Anomalies

PSAR Section 15.5.5.3 analyzes reactivity and power anomalies (all rod withdrawal and single rod withdrawal) as DEC events. However, the applicant indicated in a letter dated January 7, 2026 (TVA 2026-TN13051, non-proprietary), and in the response to RAI 15.2-8 (NRC 2026-TN13174, non-proprietary, ML26082A196, proprietary) that inadvertent single control rod withdrawal at power with DL2 CCF and all control rod withdrawal at power events will be analyzed in the FSAR as DBA event sequences instead of as DEC event sequences. The applicant's January 7, 2026, letter states that the inadvertent single rod withdrawal at power sequence to consist of an AOO postulated initiating event (inadvertent rod withdrawal) combined with failure of the DL2 mitigating functions was recategorized following PSAR analyses as part of design progression. Consistent with terminology used in the Safety Strategy methodology, this event sequence is now classified as a DBA event sequence. As described in the NRC staff evaluation of the CRN-1 event categorization scheme in Section 15.1.3.1 of this SER, the staff concluded that to meet regulatory requirements, such initiating events need to be evaluated in the FSAR assuming success of the DL2 mitigation (Type 1) and failure of the DL2 function (Type 2).

The response to RAI 15.2-8 describes that the change in classification of the inadvertent individual control rod withdrawal at power sequence requires application of a different set of analysis procedures and acceptance criteria such that acceptance criteria for the preliminary CRN-1 design would not be met. In the RAI response, the applicant stated that they are developing additional DL3 reactor trip setpoints to ensure that their acceptance criteria are not violated in an inadvertent individual control rod withdrawal at power event that is not mitigated by the DL2 automated thermal limit monitor or the multi-channel rod block. Staff considers that the results shown in PSAR 15.5.5.3 represent the nominal response of the CRN-1 system to inadvertent control rod withdrawal events, without conservatisms and considerations for uncertainty that will be applied in the final DBA analysis or the DL3 reactor trip setpoints that are under development. For the purposes of review of the CPA, staff considered the nominal system response shown in PSAR 15.5.5.3 in conjunction with the feasibility of trip setpoints discussed in RAI 15.2-8.

The applicant's proposed DL3 setpoint is based on integrated power in excess of an identified level. The NRC staff understands that a trip setpoint with this design could plausibly mitigate inadvertent individual rod withdrawal at power sequences that are limiting for the PSAR design, in which rod withdrawal leads to a slow increase in reactor power that does not reach the high-power trip setpoint. 10 CFR 50.35 permits issuance of a CP when, among other things, sufficient preliminary information is provided in the CPA and the NRC staff has determined that further information required to complete the safety analysis, which can reasonably be left for later consideration, will be supplied in the FSAR. Demonstration of the proposed DL3 setpoint to mitigate inadvertent rod withdrawal at power is needed for staff to make a final safety determination on features designed to mitigate inadvertent control rod withdrawal during the OL phase.

The staff notes that both inadvertent individual control rod withdrawal and inadvertent withdrawal of all control rods were evaluated in the PSAR. PSAR Section 7.3.3.2 notes that ganged control rod withdrawal is available in automatic and semi-automatic mode based on pre-programmed sequences. While the response to RAI 15.2-8 does not explicitly discuss categorization of ganged control rod withdrawal, staff anticipates that the same logic would apply to categorization of these faults, and therefore they would be considered in the selection of bounding reactivity insertion events for the final CRN-1 design.

In addition to inadvertent rod withdrawal at rated power conditions, the applicant also evaluated inadvertent control rod withdrawal from startup in response to RAI 15.2-8. Analysis was performed according to the methodology approved for control rod drop in NEDE-33885P-A, Revision 1, with modifications to the reactivity insertion rate made to reflect the difference in control rod speed between rod withdrawal and rod drop events and to explicitly model each channel. Staff considers explicit modeling of each channel to be acceptable for the CPA because removal of this simplification permits modeling of control rod group withdrawal, and removal of this simplification should not significantly affect results for single control rod withdrawal. The modeled control rod withdrawal speed is 28.0 mm/s, which staff confirmed to be consistent with the maximum control rod withdrawal speed provided in PSAR Table 15.5-3 consistent with the intent of L&C 1 of NEDE-33885P-A. In reviewing the preliminary analysis of this event, NRC staff considered the assumptions discussed in the RAI response.

The applicant considered faults leading to inadvertent individual and group control rod withdrawal, from withdrawal of one in-sequence rod up to simultaneous withdrawal of the **[[ ]]** four groups of control rods. While the CPA does not specify a control rod withdrawal sequence or procedure, NRC staff considers the evaluated faults to be reasonable given existing control rod withdrawal procedures. The final evaluation should address this, as well as potential faults of the rod control system, as part of the fault evaluation supporting final safety analysis consistent with the intent of L&Cs 2 and 3 of NEDE-33885P-A. The applicant considered a range of initial coolant temperatures from the minimum temperature considered in NEDE-33885P-A to saturation temperature corresponding to rated pressure. Staff finds this acceptable for the purpose of the CPA because it is a sufficiently broad range to identify the limiting initial coolant temperature. The evaluation assumed that rod withdrawal was terminated by the DL3 high neutron flux during startup setpoint, with bias applied in the conservative direction relative to the value provided in the PSAR. The applicant analyzed exposures corresponding to BOC, MOC, and EOC conditions. The staff finds evaluating a range of exposure conditions to be an acceptable approach to provide assurance that the analysis bounds mid-cycle restarts for the purposes of the CPA. The analysis assumes an initial power of **[[ ]]**; while the rod drop analysis in NEDE-33885P-A showed little sensitivity to initial power. The basis for this assumption in the rod withdrawal transient was not clear to the staff based on the PSAR and supplemental docketed information. Lower initial power levels may be more limiting for inadvertent rod withdrawal from startup conditions. The staff finds the set of assumptions and analyzed cases sufficient for the purposes of the CPA. During review of the OLA, staff will confirm that the method employed in the final analysis is appropriate for establishing the final design basis.

The applicant evaluated the results of rod withdrawal from startup with respect to the high temperature cladding failure threshold provided in RG 1.236 (NRC 2020-TN13149), Revision 0, for evaluation of control rod drop. The applicant's results show that the evaluated sequences do not exceed this threshold. The evaluation states that the PCMI failure threshold is not relevant to rod withdrawal analysis because the events involve a gradual rise in power. The applicant provided information to support this conclusion for audit, but the staff does not consider the supporting information sufficient to agree that PCMI is never relevant to rod withdrawal from startup conditions for BWRX-300. Rod withdrawal from cold conditions, in general, could result in rapid changes in power depending on the amount of reactivity added through the event and the magnitude and timing of reactivity feedback. While staff does not have enough information to conclude that BWRX-300 design would never allow this to occur such that the PCMI failure threshold is not relevant, the applicant's results for the high temperature cladding failure figure of merit show that transients evaluated for the preliminary design would not challenge this threshold. That is, the maximum enthalpy rise resulting from the applicant's preliminary analysis would not violate PCMI criterion included in RG 1.236, Revision 0 for application to rod drop. Therefore, the NRC staff considers that the question of relevance of the PCMI failure threshold to this event can reasonably be left for later consideration, in accordance with 10 CFR 50.35.

The staff concludes that the information provided describing the inadvertent single control rod withdrawal at power and all control rod withdrawal at power events, as well as the identification of safety features which are under development and description of the development program, are sufficient to assess the risk to public health and safety resulting from operation of the facility. This preliminary analysis is therefore sufficient and complies with the applicable regulatory requirements in 10 CFR 50.34(a)(4). In addition, the staff concludes that the preliminary analysis describes the principle architectural and engineering criteria for the design and has identified the major features or components incorporated therein for the protection of the health and safety of the public. The staff also concludes that such further technical and design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the FSAR. Therefore, the NRC finds the information described above supports issuance of a CP in accordance with 10 CFR 50.35.

Control Rod Drop Accident

TVA provided an alternate PDC 28 which does not require analysis of the control rod drop accident as a postulated reactivity accident. As stated in PSAR Section 15.5.5.1, analysis of the control rod drop accident as a DEC will be provided with the CRN-1 OLA. Discussion of PDC 28 and the control rod drop accident is provided in Section 4.6 of this report.

Decrease in Reactor Coolant Inventory

PSAR Section 15.5.5.4 describes a postulated event involving a spurious CCF that isolates all FW flow concurrent with a passive CCF of the safety-related DL3 functions. DL2 functions to maintain RPV pressure, initiate an anticipatory scram, and initiate ICS pressure control are credited in addition to a DL4a function to isolate MS on low FW flow. The sequence results in a decrease in reactor vessel water level and power, reactor pressure control maintaining pressure, scram and MSRIV isolation on sustained low FW flow, initiation of ICS pressure control on high RPV pressure, and achievement of a controlled state.

Because this event does not represent conditions that challenge reactivity control beyond previously evaluated transients and does not introduce new failure mechanisms or accident progression phenomena relative to other analyzed events, separate staff evaluation at the CP stage is not required by 10 CFR 50.34(a). Accordingly, the staff does not reach a safety finding on this DEC event.

15.5.5.4 Conclusion

The staff concludes that the preliminary plant design is acceptable as to transients treated as DECs. The NRC staff finds that the information in PSAR Section 15.5.5 is sufficient to describe major features and principal architectural and engineering criteria for the design and this preliminary analysis is sufficient for issuance of a CP in accordance with 10 CFR 50.34(a) and 50.35. Furthermore, the staff determined that technical and design information can be reasonably left for later consideration at the OL licensing phase and will be supplied in the FSAR, in accordance with 10 CFR 50.35. Therefore, this information adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a) and 10 CFR 50.35(a).

15.5.6 Analysis of Design Extension Conditions With Core Damage

Design Extension Conditions with Core Damage are defined in CP Section 15.1.5 as being equivalent to severe accidents. PSAR Section 15.5.6 states that the analysis of DECs with core damage are addressed in the Level 2 PSA described in PSAR Section 15.6.4, and that severe accidents identified in the Level 2 PSA will be described in the FSAR.

The staff notes that a PSA is not required to meet the applicable regulatory requirements for a CP under 10 CFR 50.33 through 50.35 and 10 CFR Part 50 (TN249), Appendix A. The staff acknowledges that the applicant included this information in its PSAR; however, the staff has not evaluated this information as part of its safety review to support the issuance of the construction permit. Accordingly, the staff does not take a position on the information presented in this section related to DECs with core damage.

15.5.7 Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool

The PSAR does not contain an analysis description of those postulated events and accident scenarios associated with the fuel pool as this will be described in the FSAR. The staff notes that this analysis is not required to meet the applicable regulatory requirements for a CP under 10 CFR 50.33 through 50.35 and 10 CFR Part 50 (TN249), Appendix A.

15.5.8 Analysis of Fuel Handling Accident

15.5.8.1 Introduction

This section of the PSAR provides a description and evaluation of the Fuel Handling Accident (FHA) for the CRN-1 design. The purpose of this section of the PSAR is to demonstrate that the design and operational features of the CRN-1 facility provide adequate protection to the public and plant personnel from potential radiological consequences associated with a postulated fuel handling accident, in accordance with the applicable regulatory requirements and guidance.

15.5.8.2 Regulatory Evaluation

The staff based its review on the relevant requirements of the following regulations and guidance:

- 10 CFR 50.34(a)(1)(ii)(D) (TN249)
- 10 CFR 50.35

- PDC 19, as it relates to the requirement that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and AOOs and to maintain it in a safe condition during design bases accidents
- RG 1.183 (NRC 2023-TN9587) Revision 1, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power,* Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident"

15.5.8.3 Technical Evaluation

The applicant categorized the FHA as a non-reactor group DBA event. The FHA is postulated to occur because of a failure of the fuel assembly lifting mechanism, resulting in the drop of a raised irradiated fuel assembly on the top of the reactor core or into the fuel pool storage racks. The dropped irradiated fuel assembly results in cladding failure in the dropped and impacted bundles and a subsequent release of fission products from the water. The applicant incorporated the methodology described in RG 1.183, Revision 1, Appendix B. Following this guidance, the applicant assumed that the chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide and that all the gap activity in the damaged rods is assumed to be released over two phases that are described below:

- Phase 1—Instantaneous release from the rising bubbles (from start of accident to 2 hours). Elemental iodine and organic iodine are conservatively assumed to be in vapor form. Elemental iodine is subsequently decontaminated by passage through the overlying pool of water into the building atmosphere while the CsI is retained by the water.
- Phase 2—Protracted release due to re-evolution as elemental iodine (starts at 2 hours and ends at 30 days). CsI is conservatively assumed to completely dissociate into the pool water. Because of the low pH of the pool water, CsI (as well as Phase 1 absorbed elemental iodine within the pool) slowly re-evolves as elemental iodine into the building atmosphere.

The NRC staff's evaluation of the radiological release consequences of the postulated FHA for the EAB, LPZ, and the Main Control Room is summarized below.

Source Term

The applicant used conservative assumptions to estimate the number of failed rods in the dropped assembly as well as failed rods in impacted assemblies. The applicant assumed that all 92 rods in the dropped assembly are assumed to fail releasing the gap activity to the pool water. For additional conservatism, the applicant assumed that all the damaged rods in the dropped assembly are full length rods. In addition, the applicant included the gap activity in a fraction of the rods in assemblies impacted from the initial drop.

The applicant used the gap fractions specified in RG 1.183 (NRC 2023-TN9587), Revision 1, including the power and burnup limitations specified in RG 1.183, Footnote 11. The applicant assumed that the reactor would be subcritical for at least 24 hours prior to the initiation of refueling. To maximize the radioactive material available for release in an FHA the applicant assumed the damaged rods are from assemblies with peak inventory using a radial peaking factor of 1.7. The applicant's results for the FHA activity released from the fuel to the pool water is shown in PSAR Table 15.5-37.

Release assumptions

The applicant analyzed the release of the activity from the pool water in two phases: (1) initial gaseous release and water depth and (2) re-evolution.

Phase 1: Initial Gaseous Release and Water Depth

The applicant states that although the depth of water is greater than 23 feet, for conservatism, the water depth was assumed to be 23 feet to allow the use of the alternate decontamination factor method described in RG 1.183, Revision. 1. Following the method described in the RG, the applicant detailed the calculation of the overall iodine decontamination factor for the Phase 1 release in Section 15.5.8 of the PSAR with the results of the Phase 1 activity released from the pool shown in PSAR Table 15.5-38a.

Phase 2: Re-evolution Phase

The staff determined that the applicant followed the guidance in RG 1.183, Revision 1, through assuming that the CsI initially released to the pool water instantaneously disassociates and dissolves in the water. Subsequently, the dissolved iodine is assumed to react chemically to form elemental iodine that is released directly to the environment over the next 30 days of the accident evaluation period. The applicant's determination of the Phase 2 initial activity available for release from the reactor cavity pool is shown in PSAR Table 15.5-38b.

Transport in the Reactor Building

The applicant followed the methodology described in RG 1.183, Revision 1, and assumed that the activity released from the reactor cavity pool mixes instantaneously with the free air volume of the refueling floor with no credit for holdup and retention within the reactor building. For the Phase 1 release rate, the applicant followed the guidance in RG 1.183, Revision 1, and used a flow rate that corresponds to the release of 99.9 percent of the activity to the environment in 2 hours. For the Phase 2 release to the reactor building, the applicant followed the guidance in RG 1.183, Revision 1, to determine the portion of the elemental iodine inventory that re-evolves due to the chemical conditions present in the water. The applicant modeled the Phase 2 release as a constant flow rate over the remaining 30 days of the accident evaluation period.

Main Control Room Model

To conservatively bound the estimated dose to the control room operators, the applicant assumed a very high unfiltered flow rate (38,500 cubic feet per minute) into the control room. The staff agrees that use of this assumption obviates the need for confirmation testing of control room unfiltered in-leakage since the dose model treats the control room as an open volume to the environment.

In a pre-application readiness assessment, the staff noted that the control room atmospheric dispersion factors (χ/Q values) appeared to be lower than the corresponding EAB values. The applicant addressed this disparity in the PSAR explaining that the offsite χ/Q values used in the dose consequence analyses were based on previously reviewed Early Site Permit (ESP) values for short-term offsite atmospheric dispersion factors calculated using the PAVAN code. After approval of the ESP, RG 1.249 (NRC 2023-TN12521), "Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors," guidance became available endorsing use of the ARCON code for calculation of EAB and the outer boundary of the LPZ dispersion factors at distances less than 1,200 meters from release source structures. The use of the ARCON code is expected to significantly reduce conservatism associated with the PAVAN code at near-field distances. In subsequent audit discussions, the applicant stated that the offsite dose consequences will be revised based on use of the more realistic ARCON atmospheric dispersion factors in the FSAR associated with the application for an OL. Since the submitted dose results in the PSAR show a considerable margin to the acceptance criteria and are expected to decrease even further using the ARCON dispersion factors for the FSAR, the NRC staff agreed that it is not necessary to revise the dose results for the PSAR.

Dose Consequence Calculation

The applicant followed all the standard assumptions in RG 1.183, Revision 1, to conservatively estimate the dose consequence from the postulated FHA. The applicant included all the key parameters used in the FHA analysis in PSAR Table 15.5-38c. The applicant's dose consequence results for the postulated FHA, which indicate a substantial margin to the acceptance criteria, are shown in PSAR Table 15.7-4.

15.5.8.4 Conclusion

The applicant evaluated the radiological consequences resulting from a postulated FHA and concluded that the radiological consequences at the EAB, LPZ, and Main Control Room are well within dose acceptance criteria of 10 CFR 50.34(a)(1)(ii)(D), and PDC 19 for control room personnel. The NRC staff finds that the applicant used analysis assumptions, inputs and acceptance criteria consistent with regulatory guidance in RG 1.183, Revision 1. The staff concludes that the EAB, LPZ, and control room doses estimated by the applicant for the FHA could meet the applicable accident dose criteria and are therefore acceptable. Therefore, the NRC staff finds that the information in PSAR Section 15.5.8 is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1) and 10 CFR 50.35, as applicable.

15.5.9 Analysis of Radioactive Releases from a Subsystem or a Component

15.5.9.1 Introduction

This section of the PSAR presents the evaluation of radioactive releases resulting from postulated line breaks in subsystems or components occurring outside containment for the CRN-1 facility. The purpose of this section is to determine whether the design features and safety analyses for the proposed design adequately address the potential consequences of postulated line breaks outside containment and to ensure compliance with applicable regulatory requirements and guidance. The evaluation is performed in accordance with the regulatory dose acceptance criteria specified in SRP Section 15.0.3, Table 1, and 10 CFR 50.34 (TN249), addressing potential exposures at the EAB, LPZ, and in the Main Control Room. The analysis demonstrates that, even under conservative assumptions, the facility's design and administrative controls are sufficient to mitigate the consequences of postulated line breaks outside containment, ensuring that offsite and onsite doses remain within regulatory limits. This section provides an overview of the accident scenarios considered, the analytical approach, and the results of the preliminary safety analysis. More detailed evaluation and final analyses will be provided in the FSAR as part of the OLA.

15.5.9.2 Regulatory Evaluation

The staff based its review on the relevant requirements of the following regulations and guidance:

- 10 CFR 50.34(a)(1)(ii)(D) (TN249)
- 10 CFR 50.35
- PDC 19, as it relates to the requirement that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and AOOs and to maintain it in a safe condition during design bases accidents
- RG 1.183, Revision 1 (NRC 2023-TN9587)

15.5.9.3 Technical Evaluation

The applicant analyzed the dose consequences following postulated breaks listed below outside containment:

- Main Steam Line Break (MSLB)
- Feedwater Line Break (FWLB)
- Isolation Condenser Line Break (ICLB)
- Instrument Line Break (ILB)

The applicant determined that no fuel damage occurs for these postulated breaks outside containment. Specifically, the applicant determined that for these postulated breaks, reactor water level will not decrease below the top of active fuel and fuel cladding temperature will not exceed the normal operating temperature. Therefore, in accordance with the guidance in RG 1.183, Revision 1, the applicant assumed the source term for these postulated events is the reactor coolant activity used for design-basis calculations with two spike cases as follows:

- Case 1—The concentration that is the maximum value permitted under plant operating limits and conditions (typically 4.0 microcuries per gram dose equivalent iodine-131) and corresponds to the conditions of an assumed pre-accident spike.
- Case 2—The concentration that is the maximum equilibrium value permitted under plant operating limits and conditions for continued full-power operation (typically 0.2 microcuries per gram dose equivalent iodine-131).

The staff determined that the applicant followed the guidance in RG 1.183, Revision 1, to conservatively estimate the control room and offsite dose consequences from these postulated events. From a dose consequence perspective, the only significant difference in these events is the quantities of reactor water and steam released. For the MSLB, FWLB, and the ICLB, the plant is designed to immediately detect such occurrences and initiate isolation of the affected lines, thereby limiting the releases. The applicant's results for the activity released for the MSLB, FWLB, and ICLB are shown on PSAR Table 15.5-39, Table 15.5-40, and Table 15.5-41, respectively.

For the ILB, the applicant assumed that the release is not automatically isolated and operator action to initiate a controlled shutdown is credited to occur after 72 hours. The controlled shutdown is assumed to occur over a 5.2-hour period. Therefore, the release is assumed to occur over a 77.2-hour period. The activity released for the ILB over the assumed release period is shown for the equilibrium and the pre-accident spike source terms in PSAR Table 15.5.42a and Table 15.5.42b, respectively.

Transport

The staff determined that, consistent with RG 1.183, Revision 1, the applicant assumed that all activity is released from line breaks outside containment without credit for plate out, holdup, or dilution within facility buildings.

Main Control Room Model

The applicant modeled the control room using the same conservative assumptions used for the FHA, which conservatively bounded the estimated dose to the control room operators assuming a very high unfiltered flow rate (38,500 cubic feet per minute) into the Main Control Room. The staff determined that the use of this assumption obviates the need for confirmation testing of control room unfiltered in-leakage since the dose model treats the control room as an open volume to the environment.

Dose Consequence Calculation

Based on the information discussed above, the staff determined that the applicant followed all the standard assumptions in RG 1.183, Revision 1, to conservatively estimate the dose consequences from postulated line breaks outside containment. The applicant's dose consequence results for the MSLB, FWLB, ICLB, and ILB indicate substantial margins to the acceptance criteria as shown in PSAR Table 15.7-5, Table 15.7-6, Table 15.7-7, and Table 15.7-8.

15.5.9.4 Conclusion

The applicant evaluated the radiological consequences resulting from postulated line breaks outside containment and concluded that the radiological consequences at the EAB, LPZ, and control room are well below the dose acceptance criteria of 10 CFR 50.34(a)(1)(ii)(D) and PDC 19 for control room personnel. The NRC staff finds that the applicant used analysis assumptions, input, and acceptance criteria consistent with regulatory guidance in RG 1.183, Revision 1. The staff concludes that the EAB, LPZ, and control room doses estimated by the applicant for postulated line breaks outside containment could meet the applicable accident dose criteria and are therefore acceptable. Therefore, the NRC staff finds that the information in PSAR Section 15.5.9 is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a)(1) and 10 CFR 50.35, as applicable.

15.6 <u>Probabilistic Safety Assessment</u>

15.6.1 Introduction

Section 15.6, "Probabilistic Safety Assessment," of the Clinch River PSAR summarizes the applicant's PRA. As described in Section 15.6.10, "Probabilistic Safety Assessment Insights and Applications," PRA activities are ongoing and will continue to mature as design progresses.

As discussed in PSAR Sections 15.2.1 and 15.2.2, the PRA is intended to play a central role in the Chapter 15 accident analyses by providing quantitative frequencies for event sequences, which would be categorized within each fault group as AOOs, DBAs, or DECs, with exceptions noted in NEDC-33934P. The PRA also is intended to support sequence selection (including the complex sequence selection and severe accident sequence selection described in PSAR Section 15.2), complement the deterministic safety analysis (PSAR Section 15.1.1), and confirm that the safety goals for CDF and LRFs are met (PSAR Section 15.3.2). The PSAR further describes the iterative design process through which PRA insights inform design decisions, consistent with the applicant's stated risk-informed design approach (PSAR Sections 15.6.1, 15.6.1.1, and 15.6.2.2). However, as stated in PSAR Section 15.2.2 (Supplement 6, February 4, 2026, A-15.2-3 response), the applicant indicated that qualitative analyses are used to categorize events in the PSAR in support of the CP and that the FSAR, quantitative frequencies based on Level 1 PSA results will be adopted as the final, governing measure for event sequence categorization.

In addition, as indicated in PSAR Section 15.5.6, DEC sequences resulting from complex sequence selection and severe accident sequence selection are not submitted for the PSAR to support the CP review; they will be included in the FSAR.

The staff notes that the discussion and review of the use of event sequence frequencies for categorization of AOOs, DBAs, and DECs is provided in Section 15.1.3.1 of this report.

15.6.2 Regulatory Evaluation

The staff determined that a detailed safety evaluation is not needed at this stage to meet the requirements of 10 CFR 50.34 (TN249) for a CPA.

Under 10 CFR 50.34, "Content of applications; technical information," an applicant for a CP is required to submit a preliminary safety analysis report (PSAR) that includes certain specified information. The regulation identifies the minimum information to be included in the PSAR but does not require the development or submission of a PRA.

While 10 CFR Part 50 does not require development of a PRA for a CPA, a systematic assessment of plant risk, including use of PRA insights, may help demonstrate that the design, construction, and operation of the reactor will reflect an extremely low probability of accidents resulting in significant releases of radioactive fission products, consistent with 10 CFR 50.34(a)(1)(ii) and applicable Commission policy statements.

15.6.3 Technical Evaluation

The staff acknowledges the applicant's stated intent is to further develop and refine the PRA and associated risk insights as the design matures and to incorporate those results in the FSAR. The staff will review those analyses at the FSAR stage to verify conformance with applicable Commission policy statements and regulatory guidance related to the use of risk information, configuration control, and peer review of PRA models. Consistent with this approach, the applicant has noted in PSAR Table 1.9-19 that conformance with SRP Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," will be demonstrated at the FSAR.

15.6.3.1 PRA Objectives and Scope

Section 15.6.10 and Table 15.6-1 of the PSAR identify the PRA objectives, which are listed below:

1. providing a systematic analysis to demonstrate protection of the public and the environment
2. demonstrating a balanced design such that no single feature or PIE dominates the risk profile
3. providing confidence that cliff-edge effects are prevented
4. assessing quantitative safety goals (core damage and release frequencies)
5. performing site-specific assessments of external hazards
6. identifying vulnerabilities and potential design or procedural improvements
7. assessing adequacy of emergency operating procedures
8. providing insights for severe accident management

The applicant stated that objectives 1–6 are being addressed during the current design stage, while objectives 7–8 will be assessed once detailed design and operating procedures are developed.

The PSAR describes the PRA models developed by the applicant to encompass all modes and hazards, including internal events, internal flooding, internal fires, seismic events, and high winds. These models are intended to address all modes, all internal and external events, and combinations of hazards where applicable. The methodology used to develop and apply the PRA models also is described in the PSAR.

15.6.3.2 Status of the PRA and Related Activities

PSAR Section 15.6.1 explains that the PRA, together with the deterministic safety analysis, supports the applicant's understanding of overall risk and dominant contributors.

The PRA models and results described in the PSAR are preliminary. The applicant stated that:

- The PRA has not yet undergone an independent peer review. A full-scope self-assessment and peer review against the endorsed PRA standard (RG 1.200) will be performed to support the FSAR (Section 15.6.2).
- PRA configuration control measures will be described in the FSAR (PSAR Section 15.6.2).
- DEC sequences involving core damage and severe accidents will be included in the FSAR (PSAR Sections 15.1.5 and 15.5.6).

15.6.3.3 Staff Observations

At this stage of design development, the staff reviewed the applicant's description of the PRA and its integration with the design process. The staff notes the following:

- The PRA is intended to meet the applicable Commission policy statements and regulatory guidance on PRA quality, including RG 1.200 and the Commission Policy Statement on the Use of PRA Methods in Nuclear Regulatory Activities.

- The applicant has stated its intent to conduct a peer review of the PRA prior to FSAR submittal and to maintain configuration control in accordance with established standards.
- The applicant-reported PRA results provided preliminary estimates of CDFs and LRFs, supporting the applicant's expectation that the safety goals will to be met. However, the staff cannot confirm this for the CP because DEC sequences resulting from complex sequence selection and severe accident sequence selection (PSAR Section 15.5.6), as well as descriptions of DL4 design features credited for prevention or mitigation of core damage and severe accidents (Appendix 5B) were not submitted or reviewed.

Given the above, the staff takes no position regarding the adequacy of the PRA or confirming the safety goals are met.

15.6.4 Conclusion

The staff recognizes that the PRA described in the PSAR is still being refined and will continue to evolve with design maturity. Given that the design and supporting PRA models are still being developed, the PRA has not yet undergone an independent peer review, and the severe accident and beyond-design-basis evaluations are deferred to the FSAR, the staff does not take a position on the adequacy of the PRA at the CP stage.

At the FSAR stage, the staff will review the completed PRA, including its peer review results, configuration control process, and incorporation of all internal and external hazards, to determine whether it meets the applicable regulatory requirements and supports the design's demonstration of compliance with quantitative safety goals and meeting the stated PRA objectives, such as demonstrating a balanced design, providing confidence that cliff-edge effects are prevented, and identifying vulnerabilities and potential design or procedural improvements.

15.7 Results of the Deterministic Safety Analyses and Probabilistic Safety Assessment

This section of the PSAR provides the results of the analyses described in Sections 15.5 and 15.6. See Sections 15.5 and 15.6 of this report for the staff's evaluation.

15.8 Appendix 15A Practical Elimination Claims and Provisions

This appendix to the PSAR provides a tabular summary of the applicant's initial identification of events or sequences considered for practical elimination. The applicant states in Section 15.1.6 that the practically eliminated event sequences with core damage will be finalized and provided in the FSAR.

The staff notes that the information in this section is not required to meet the applicable regulatory requirements for a construction permit under 10 CFR 50.33 through 50.35 and 10 CFR Part 50 (TN249), Appendix A. The staff acknowledges that the applicant included this information in its PSAR; however, the staff has not evaluated this information as part of its safety review to support the issuance of the CP. Accordingly, the staff does not take a position on the information presented in this section related to practical elimination claims and provisions.

15.9 Appendix 15B Complementary Defense Line 4 Functions for Mitigating DECs

Appendix 15B to the PSAR identifies complementary design features intended to prevent accident progression or mitigate the consequences of DECs. The appendix states that these features and their mitigating functions will be finalized as the beyond-design-basis accident Level 2 PSA analysis progresses. The applicant states in Section 15.5.7 that severe accidents will be described in the FSAR.

The staff notes that the information in this section is not required to meet the applicable regulatory requirements for a CP under 10 CFR 50.33 through 50.35 and 10 CFR Part 50 (TN249), Appendix A. The staff acknowledges that TVA included this information in its PSAR; however, the staff has not evaluated this information as part of its safety review to support the issuance of the CP. Accordingly, the staff does not take a position on the information presented in this section related to complementary design features for mitigating DECs.

15.10 Appendix 15C Evaluation of Radiological Consequences Per 10 CFR 50.34(a)(1)(ii)(D)

15.10.1 Introduction

This section of the PSAR provides a description and evaluation of the MHA for the CRN-1 BWRX-300 design. The fission product release assumed for this evaluation should be based on a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. The purpose of this section is to demonstrate that the design and operational features of the CRN-1 facility provide adequate protection to the public and plant personnel from the potential radiological consequences associated with an MHA, in accordance with the applicable regulatory requirements and guidance.

15.10.2 Regulatory Evaluation

The staff based its review on the relevant requirements of the following regulations and guidance:

- 10 CFR 50.34(a)(1)(ii)(D) (TN249)
- RG 1.183, Revision 1, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” (NRC 2023-TN9587)

15.10.3 Technical Evaluation

The applicant submitted an evaluation assuming a fission product release from the core into containment to satisfy the requirements of 10 CFR 50.34(a)(1)(ii)(D). The applicant based their evaluation on an MHA that results in core melt consistent with RG 1.183, Revision 1 (NRC 2023-TN9587). The applicant correctly described the MHA as a conservative surrogate accident that is intended to challenge aspects of facility design to demonstrate the ability of the facility to limit the offsite and onsite consequences of such an event. In addition to the summary of the MHA evaluation contained in PSAR Appendix 15C, as an element of the audit process, the staff audited the proprietary MHA calculation in the electronic reading room.

In conformance with 10 CFR 50.34, Footnote 3, the applicant's MHA evaluation assumes a substantial meltdown of the core with a subsequent release into containment of appreciable quantities of fission products. The applicant stressed that their MHA evaluation was neither derived from a PRA fault evaluation process nor is it categorized by the BWRX-300 Safety Strategy. Rather, the applicant correctly described their MHA evaluation as a design-basis hypothetical major accident involving a substantial core melt release from the core to the containment to assess the performance of containment and systems that mitigate the potential release of fission products in accordance with 10 CFR 50.34(a)(1)(ii)(D).

The applicant analyzed the MHA consequences in terms of the TEDE at the EAB and the outer boundary of the LPZ. In addition, the applicant evaluated the potential TEDE dose in the MCRE against the dose criterion of PDC 19.

The applicant stated that multiple active failures are assumed to hypothesize the MHA and that because the BWRX-300 design does not use active containment fission product removal systems to mitigate the radiological consequences, assuming an additional single active component failure that results in the most limiting radiological consequences is not postulated. The NRC staff agrees that for the BWRX-300 design, which does not rely on active containment fission product removal systems to mitigate the radiological consequences of an accident, assuming an additional single active component failure is not necessary for the MHA evaluation.

Chapter 6 in the PSAR states that the Engineered Safety Features (ESFs) for the BWRX-300 design are passive systems and are not dependent on external sources of power or operator action to fulfill the Fundamental Safety Functions for at least 72 hours after a DBA. The NRC staff notes that the MHA analysis credits SC1 systems to provide containment isolation as well as SC3 systems for control room isolation, pressurization, and filtration. Systems classified SC1 are powered by DC batteries with a coping time of at least 72 hours without operator action or AC power. These SC3 systems have backup AC power from Standby Diesel Generators in the event of a loss of offsite power. Therefore, the NRC staff concludes that the applicant's MHA analysis remains valid should a loss of offsite power occur.

The applicant used the following computer codes to perform the MHA analysis:

- The applicant used the ARCON code, NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," May 1997, to model radionuclide transport to the onsite MCRE and offsite EAB and LPZ locations. The ARCON code has been endorsed by the NRC for determining atmospheric relative concentration (χ/Q) values, also known as atmospheric dispersion factors, in support of modeling releases from a DBA.
- The applicant used the Modular Accident Analysis Program (MAAP) code, "MAAP 5.06 Applications Guidance," Electric Power Research Institute, September 2021, to model the progression of an MHA and model fission product transport and natural deposition in the primary containment system (PCS). The applicant used the MAAP code in its MHA analysis to quantify fission product removal mechanisms for isotopes in aerosol form. The MAAP code is widely used by nuclear utilities and research organizations to predict the progression of LWR accidents.
- The applicant determined the dose consequences of a postulated MHA using the RADTRAD Version 3.10 computer code as described in NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation." The RADTRAD code was developed by the Accident Analysis and

Consequence Assessment Department at Sandia National Laboratories for the NRC, Office of Nuclear Reactor Regulation, Division of Reactor Program Management. If deemed necessary to make a reasonable assurance finding, the staff can perform independent confirmatory radiological calculations using the RADTRAD computer code, run within the SNAP suite of integrated applications for engineering analysis, developed for the NRC. Information on the SNAP/RADTRAD code is available from the NRC's Radiation Protection Computer Code Analysis and Maintenance Program at https://ramp.nrc-gateway.gov/content/snapradtrad-overview.

- The applicant used the MicroShield code, "MicroShield Version 7, User's Manual," Grove Engineering Inc., 2009, to calculate the MHA direct shine dose to the MCRE. The MicroShield code is widely used by the nuclear industry for calculation of direct shine dose.

Source term:

In conformance with RG 1.183, Revision 1, the applicant evaluated the inventory of fission products in the reactor core based on the maximum expected core thermal power for an assumed 24-month refueling cycle maximizing the fission product inventory. To account for core power measurement uncertainty the applicant applied a factor of 1.02 to the calculated core inventory.

In conformance with RG 1.183, Revision 1, the applicant assumed that the radioactivity released from the fuel in the MHA mixes instantaneously and homogeneously throughout the free air volume of the PCS as it is released. The applicant conservatively assumed that the gap release phase begins at the onset of the MHA. Consistent with RG 1.183, Revision 1, the applicant modeled the activity released from the core during each release phase as increasing in a linear fashion over the duration of the phase. The applicant incorporated the core inventory release fractions and MHA timing from RG 1.183, Revision 1, Tables 1 and 5, respectively.

The applicant's MHA evaluation assumed that the PCS sump pH is 7 or greater for 30 days and stated that this assumption will be verified by design or analysis prior to the FSAR. The applicant's MHA calculation identified the assumption that the PCS sump pH is maintained to a value of at least 7 as an open item. Consistent with RG 1.183, Revision 1, Assumption A-1.1, the applicant assumed the chemical form of radioiodine released to the PCS atmosphere to be 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic iodide.

The applicant credited natural deposition on containment surfaces and the retention in the pool of water that will accumulate at the bottom of containment as a result of the MHA. The applicant credited the removal of elemental iodine and aerosol fission products by deposition on containment surfaces for all leakage pathways. The applicant did not credit the removal of organic iodide in the PCS for the MHA evaluation.

Because the BWRX-300 does not contain containment sprays, the only elemental iodine removal credited by the applicant is deposition on PCS surfaces. The applicant calculated an elemental iodine removal rate constant based on guidance found in NUREG/CR-0009, "Technical Bases for Models of Spray Washout of Airborne Contaminates in Containment Vessels," NRC, October 1978, as referenced in SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," (NRC 2007-TN13176). The NRC staff notes that while elemental iodine was assumed to be the dominant iodine species in post-accident containment atmospheres under the assumptions of the original fuel melt source term described in Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test

Reactor Sites," March 23, 1962; under assumptions of the alternative source term, CsI as an aerosol replaced elemental iodine as the dominant iodine species in post-accident containment atmospheres.

The applicant credited elemental iodine removal by deposition on the containment wall surface area, 50 percent of the containment floor area as well as portions of the PCCS external pipe surface area. As shown in PSAR Table 15C-5, the applicant employed a containment elemental iodine removal coefficient of 1.69 per hour for 9.59 hours. The applicant limited the credit for elemental iodine removal by natural deposition to a factor of 200 which occurs at 9.59 hours.

The applicant determined the fraction of fission products in aerosol form removed from the containment atmosphere by natural processes using the MAAP code. To determine the thermal and hydraulic conditions for aerosol deposition, the applicant modeled the MAAP code scenario as a double-ended guillotine break in an ICS return line resulting in the core exposure and melting. Following the guidance in RG 1.183, Revision 1, the applicant terminated the release into containment at the end of the early in-vessel release phase. The applicant provided the time-dependent aerosol removal rates in PSAR Table 15C-2. The applicant's MHA calculation identified the **[**

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Transport:

The applicant postulated leakage from the PCS through the following pathways:

- containment leakage into the Reactor Building (RB)
- bypass leakage that bypasses the RB
- ESF leakage

In accordance with guidance in RG 1.183, Revision 1, the applicant evaluated leakage from the PCS to the RB at the design leakage rate of 0.35 weight percent per day. After 24 hours, the applicant reduced the assumed leakage by 50 percent in accordance with RG 1.183, Revision 1. As described in PSAR Section 9.4.6, the RB heating and ventilation system (HVS) isolates automatically using Safety Class 3 (SC3) isolation dampers providing containment and confinement to limit releases of airborne radioactivity to the environment under postulated accident conditions. The applicant did not credit RB dilution or holdup when evaluating the offsite dose consequence. The applicant did credit dilution and holdup of containment leakage into the RB for calculating onsite dose. The applicant's analysis does not take credit for any operator actions to mitigate the radiological consequences of the postulated MHA.

The applicant conservatively modeled leakage into the RB using only the portion of the RB that is above grade. Following the guidance in RG 1.183, Revision 1, the applicant limited the mixing credited for calculating onsite dose to 50 percent of the free air volume in the above grade portion of the RB. The applicant did not credit the deposition of fission products on RB surfaces. The applicant assumed that after automatic isolation the RB exfiltration rate is 80 cubic feet per minute which equates to 54.6 weight percent of the free air mixing volume each day. The applicant's MHA calculation identified **[[**

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The applicant evaluated the potential for leakage through containment isolation valves (CIVs) in lines that lead directly to the environment or to buildings other than the RB. For these pathways, the applicant credited holdup in the PCS in addition to elemental iodine and aerosol fission product removal by deposition within containment. The applicant did not take any additional deposition credit in the individual lines. The applicant credited holdup in larger pipes; however, the applicant did not credit holdup for smaller lines. The applicant grouped smaller lines together as "generic bypass lines."

To credit holdup in the larger RB bypass lines the applicant tabulated the individual piping dimensions. The analysis determined the volumes for each line as well as the estimated flow rates based on the apportioned amount of the total bypass leakage. This method enabled modeling each individual bypass line as a separate compartment in the RADTRAD model.

The applicant modeled leakage from Main Steam Lines (MSLs) A and B, that extend from the RPV through the RIVs and the RB and into the Turbine Building (TB). The applicant modeled leakage from the MSL Train A outboard CIV as the sum of the leakage from MSL A, the RPV Head Vent, and the ICS Gas Purge Line A. The ICS Gas Purge Lines B and C form a pathway from ICS Condenser B and C, respectively, to MSL Train B. The applicant modeled MSL Train B leakage as the sum of the MSL B, ICS Gas Purge Line B, and ICS Gas Purge Line C leakage rates.

The applicant described the CRD as containing two CIVs associated with the purge and charging water lines for each of the Hydraulic Control Units. Piping connections are provided to the Condensate and Feedwater Heating System (CFS) in the TB. The applicant modeled 29 Hydraulic Control Units with a conservative piping configuration.

The applicant described the CUW as providing a leakage pathway to the TB. The CUW has two trains, A and B, from the RPV, with two sets of RIVs. Because the two lines join prior to the containment penetration, the applicant modeled the CUW as a single train.

The applicant described the CFS as containing four FW lines from the RPV, with two RIVs per line. The four lines join into two lines prior to the containment penetration. Therefore, the applicant modeled FW leakage as two trains, each accounting for two FW lines.

The applicant described the CIS as containing three lines with potential CIV leakage. The CIS supply, the CIS exhaust, and the CIS overpressure vent lines each provide a separate release point outside of the RB.

To account for different potential release points the applicant modeled two generic bypass lines as follows:

- Generic Train A represents releases on the RB/TB interface (e.g., the Plant Pneumatic System).
- Generic Train B represents releases into the Radwaste Building, including the Liquid Waste Management System, Demineralized Water, and the Equipment and Floor Drain System.

In accordance with RG 1.183, Revision 1, the applicant's MHA analysis includes an assessment of leakage from ESF systems that recirculate sump water outside of the primary containment during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems.

The applicant stated that since the BWRX-300 containment does not include a suppression pool, the only contributor to ESF leakage is from the ICS during the recirculation of reactor coolant outside the primary containment. The applicant assumed that ESF leakage occurs from the ICS condensate return lines.

The applicant stated that since ESF seat leakage is included in the design leakage rate, only ESF seal leakage is evaluated following the assumptions in RG 1.183, Revision 1. The applicant credited RB mixing and holdup for ESF leakage into the RB. The applicant included a correction for flashing and density for all ESF leakage to account for the leakage being defined as liquid leakage but modeled as steam leakage. The NRC staff finds this acceptable because it is consistent with the guidance in RG 1.183, Revision 1.

Control room habitability:

The applicant's MHA analysis credits the Control Building (CB) HVS for reducing airborne radiological exposure to personnel located in the MCRE. The applicant assumed that within 10 minutes of the initiation of the MHA the CB HVS would switch from normal to SC3 emergency filtration. The CB HVS is described in PSAR subsection 9.4.1 and states that MCRE emergency filtration units operate automatically upon detection of a high radiation level at the CB air handling unit outside air intake. The emergency filtration units provide a filtered intake flow into the MCRE, thus pressurizing the MCRE with respect to adjacent spaces to maintain Main Control Room habitability during a postulated MHA.

The applicant's MHA calculation describes the control room envelope emergency filtration unit as consisting of two high-efficiency particulate air filters and one charcoal bed filter. The applicant's MHA calculation **[[**

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The applicant's MHA calculation identified the **[**

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Following the guidance in RG 1.183, Revision 1, the applicant's MHA analysis considered the following potential sources of radiation exposure to the MCRE:

1. Contamination by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility.
2. Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope. The applicant states that since the MCRE is a separate structure maintained at a positive pressure, infiltration from adjacent structures to the MCRE is not expected to be significant.
3. Radiation shine from the external radioactive plume released from the facility.

4. Radiation shine from radioactive material in the containment.
5. Radiation shine from radioactive materials in systems and components inside or external to the MCRE (e.g., radioactive material buildup on MCRE filters).

Following the guidance in RG 1.183, Revision 1, the applicant considered a hypothetical maximum exposed individual who is present in the control room for 100 percent of the time during the first 24 hours after the event, 60 percent of the time between 1 and 4 days, and 40 percent of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual is assumed to be 3.5×10^{-4} m^3/s. For additional conservatism, the applicant increased the calculated MCRE dose by 10 percent to account for uncertainties in the evaluation.

Radionuclide Dispersion:

The applicant evaluated radionuclide transport from the potential sources to onsite and offsite locations based on atmospheric dispersion factors (χ/Qs) calculated for CRN-1. Following the guidance in RG 1.183, Revision 1, the applicant did not credit depletion during transport to either onsite or offsite dose locations. Following RG 1.183, Revision 1, Regulatory Position 5.3, the applicant assumed that the most limiting 2-hour EAB χ/Q value occurs simultaneously with the limiting release to the environment.

Onsite Radionuclide Transport:

For conservatism, the applicant modeled the maximum MCRE χ/Q for either the normal or emergency air intake location for the MCRE dose analysis. For leakage from the RB, the applicant modeled the release as a ground-level diffuse area source consistent with RG 1.194 (NRC 2003-TN13306), "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." NRC, June 2003. The applicant modeled RB bypass leakage as a ground-level release direct to the environment from two release points. For leakage at the TB/RB boundary, the applicant assumed that the release point is at the steam tunnel. For all other bypass leakage, the applicant assumed the release occurs at the limiting RB/CB boundary.

To ensure a conservative dose analysis, consistent with Position 5.3 of RG 1.183, Revision 1, the applicant assumed that the period of the most adverse release of radioactive materials to the environment occurs coincidently with the period of most unfavorable atmospheric dispersion. The applicant provided the onsite dispersion factors in PSAR Table 15C-3.

The NRC staff reviewed the key assumptions as presented in PSAR Section 15C.2.4.1 and the results provided in PSAR Table 15C-3. The applicant provided a detailed explanation as part of the PSAR audit material, including details on the use of ARCON for determining the MCRE dispersion factors. The staff was able to confirm the applicant's values and therefore find the control room envelope dispersion factors to be acceptable.

Offsite Radionuclide Transport:

The applicant followed the guidance in RG 1.249 (NRC 2023-TN12521), Revision 0, "Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors" NRC, August 2023, to calculate offsite χ/Qs at the EAB and LPZ locations. RG 1.249 provides guidance for using the ARCON code to estimate χ/Qs out to distances of 1,200 meters from the nearest edge of a building. The applicant stated that because the χ/Qs established for the CRN-1 Site in the “Clinch River Nuclear Site Early Site Permit, ESP-006” (NRC 2019-TN10325) were determined prior to the issuance of RG 1.249, the applicant recalculated χ/Qs for the MHA analysis at the EAB distance (335 meters) and conservatively applied these values to the LPZ.

The applicant assumed the period of the most adverse release of radioactive materials to the environment occurs coincidently with the period of most unfavorable atmospheric dispersion. The applicant noted that the worst 2-hour period for offsite doses may be different than for onsite dose because the applicant credits mixing and holdup in the RB for evaluating the onsite dose. The applicant provided the offsite dispersion factors in PSAR Table 15C-4.

The NRC staff reviewed the key assumptions as presented in PSAR Section 15C.2.4.2 and the results provided in PSAR Table 15C-4. The applicant provided a detailed explanation as part of the PSAR audit material, including details on the use of ARCON for determining the EAB/LPZ dispersion factors. The staff was able to confirm the applicant’s values and therefore find the EAB/LPZ dispersion factors to be acceptable.

15.10.4 Conclusion

The NRC staff reviewed the key assumptions as presented in PSAR Table 15C-5, the methodology employed by the applicant as described in PSAR Appendix 15C as well as the applicant’s MHA calculation. The MHA calculation identified several open items that will need to be verified prior to the issuance of an OL. The applicant’s preliminary results, as shown in PSAR Table 15C-6, could meet the acceptance criteria in 10 CFR 50.34(a)(1)(ii)(D) and PDC 19. Therefore, the NRC staff finds that the applicant’s MHA dose consequence analysis is sufficient to enable granting a CP for the CRN Site in accordance with 10 CFR 50.34(a) and 10 CFR 50.35.

15.11 BWRX-300 TRACG Application

15.11.1 Introduction

An overview of the hydraulic and heat transfer analysis methods are specified in the PSAR, with detailed descriptions of these methods provided in referenced documents. The application of those methods to the BWRX-300 design are provided in NEDC-34043, “BWRX-300 TRACG Application” (GE Hitachi 2025-TN13070, non-proprietary; ML25141A241, proprietary) (“BWRX-300 TRACG Application”). The BWRX-300 TRACG Application technical report provides a description of how TRACG, combined with other codes, is applied, and why it is qualified to perform BWRX-300 calculations for LOCA and non-LOCA events. It provides an overview of the historical development and code qualification of TRACG for operating reactors, including the SBWR and ESBWR designs, with references to supporting documents. It also describes how TRACG, with links to other codes, will be used for the BWRX-300 Chapter 15 analysis for AOO, DBA, and DEC events that do not involve core damage (events that are not severe accidents). This technical report has not previously been reviewed and approved by the staff.

15.11.2 Regulatory Evaluation

Title 10, Part 50, of the Code of Federal Regulations (10 CFR Part 50-TN249), “Domestic Licensing of Production and Utilization Facilities,” Section 50.34(a), “Preliminary safety analysis report” (10 CFR 50.34(a)(1), (4) and (5), specify, in part, the following requirements regarding applications for construction permits:

1. Safety analysis reports must analyze the design and performance of SSCs, and their adequacy for the prevention of accidents and mitigation of the consequences of accidents.
2. Analysis and evaluation of ECCS cooling performance following postulated LOCAs must be performed in accordance with the requirements of 10 CFR 50.46.
3. An identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design.

10 CFR 50.35(b), “Final safety analysis report,” includes parallel provisions for the OL stage in 10 CFR 50.35(b)(2), (4), and (6)(vi). Technical specifications for the facility are based on 10 CFR 50.36. The analyses needed for the requirements above are performed with an evaluation model. RG 1.203, “Transient and Accident Analysis Methods,” provides guidelines for how to develop an acceptable evaluation model that will be used for Chapter 15 transient and accident analyses. The PSAR states that it conforms with RG 1.203 in Table 1.9-20. RG 1.157 (NRC 1989-TN13259), “Best-Estimate Calculations of Emergency Core Cooling System Performance” describes models, correlations, data, model evaluation procedures, and methods that are acceptable-to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance during a LOCA and for estimating the uncertainty in that calculation. PSAR Table 1.9-20 also states that it commits to only the RG 1.157 guidance relating to the “general best practice elements of a rigorous method development” are applicable for the approved containment methodology (NEDC-33922P-A). 10 CFR 50, Appendix K describes the required and acceptable features for Appendix K compliant ECCS Evaluation Models. It also describes the documentation requirements for ECCS evaluation models. SRP Section 15.0.2, “Review of Transient and Accident Analysis Methods,” describes how evaluation models should be reviewed. In PSAR Table 1.9-15, the applicant states that the CPA conforms with SRP Section 15.0.2.

15.11.3 Technical Evaluation

PSAR Implementation and Analysis of BWRX-300 TRACG Application

The staff reviewed how the BWRX-300 TRACG Application was implemented in the Chapter 15 PSAR safety analysis. TRACG is a general-purpose systems code that has been used as the basis for approved EMs used for safety analysis in operating reactors, ESBWR, and BWRX-300. It is combined with other computer codes for some of the EMs. BWRX-300 TRACG Application describes how TRACG, combined with other codes, is applied and why it is qualified to perform BWRX-300 calculations for LOCA and non-LOCA events. Several approved methodologies were discussed in BWRX-300 TRACG Application, showing that it does have a pedigree and extensive validation to perform TH system safety analysis for operating BWRs, the ESBWR, and BWRX-300 (containment methodology), but of all of the approved methodologies, only the containment methodology is used for the BWRX-300 Chapter 15 PSAR analyses.

An EM includes more than a computer code like TRACG. An EM is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or DBA. As described in RG 1.203 (NRC 2005-TN13304), the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event including:

1. procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation)

2. specification of those portions of the analysis not included in the computer programs for which alternative approaches are used

3. all other information needed to specify the calculational procedure

The BWRX-300 TRACG Application scope includes Normal Operation (Stability Application), AOO BL-DSA, DBA BL-DSA, DBA CN-DSA, and EX-DSA or DEC where BL, CN, and EX signify baseline, conservative, and design extension, respectively. The acceptance criteria for each type of analysis are given. Stability calculations are required to show that the core decay ratio is less than ~0.8. The AOO BL-DSA calculations must meet MCPR derived acceptance criteria, fuel melt and cladding strain criteria, and RCPB acceptance criteria. Only rated power and nominal initial plant conditions are used in AOO calculations. The BWRX-300 TRACG Application report states that setpoints and other plant performance parameters, such as valve stroke times, are treated in a conservative way. The staff notes that at the OL phase, the applicant will need to demonstrate that these parameters are treated appropriately. DBA BL-DSA are performed in the same way with PCT and cladding oxidation replacing MCPR, fuel melting, and clad strain as the acceptance criteria. The acceptance criteria applied to AOO and DBA analysis are described in more detail in Section 15.3 of this report. The baseline analyses use nominal initial conditions and best estimate assumptions, therefore uncertainties are not accounted for in BL-AOO and BL-DBA calculations. The DBA CN-DSA adds dose acceptance criteria and uses conservative initial conditions in addition to setpoints and other plant performance parameters, such as valve stroke times, in a conservative way. BWRX-300 TRACG Application report states that DBA CN-DSA uncertainties are addressed with a graded approach depending on margin to the derived acceptance criteria. It should be noted that it is not possible to quantify if a margin is large, medium, or small if the uncertainty in the calculation has not been evaluated. GVH has not provided quantitative criteria that can be used to evaluate if an event is a high, medium, or low margin event. The staff notes that in interaction with the applicant's vendor, the PSAR arguments about the margin centered on engineering judgement associated with the preliminary design and might not be what will be done for the OLA analyses.

For high margin or substantially non-limiting events, a nominal calculation is performed. Although GVH includes conservative assumptions for parameters such as valve closure times, trip setpoints, and scram times, there is not an actual operating BWRX-300 plant design with operating experience to provide a basis for these assumptions. Therefore, the staff views these assumptions as design specifications that must be justified at the OL phase. The calculations use nominal initial conditions, rather than considering all potential initial conditions within the normal operating bands. Therefore, it is possible that the calculations may be non-conservative since there is no evaluation of possible biases in the baseline modeling.

For medium margin events, important parameters are biased by one or two standard deviations in the conservative direction. However, there is no documented method for deciding if the parameters should be biased by one or two standard deviations, and there is no demonstration to show that the results are more conservative than the results of a full uncertainty analysis. This method has been applied to a limited number of DBAs.

For low margin events, a quantitative uncertainty analysis is performed. However, this option has not been applied in the PSAR.

The AOO events were all considered to be large margin or substantially non-limiting events and were performed with only nominal initial conditions, including reactor power. Many of the DBA events were also analyzed using nominal initial conditions. It is noted that one event, the limiting DBA turbine trip/load rejection event, had a boiling transition and fuel cladding heat-up. This event is considered to be a medium margin event that used biasing on relevant PIRT multipliers. However, a boiling transition and fuel cladding heat-up does not typically occur in calculations using the approved TRACG for AOO methodology when it was used for DBA transients nor in older approved methodologies used for operating plants. If such an event did occur, it would be tabulated as a fuel failure, and a dose consequence analysis would need to be performed. In this method, the fuel cladding heat-ups are not categorized as automatic fuel failures for non-LOCA DBAs. GVH's position is that the TRACG fuel rod model can calculate if fuel failures occur in transients, but validation for this capability in non-LOCA transient calculations has not been provided in the TRACG documentation. TRACG has also not been shown to give equivalent results to the PRIME fuel rod model that is used in the approved methodology based on TRACG and TRACG for AOOs. The only mechanistic fuel failure predictions by the TRACG fuel rod model that have been approved in the past are for LOCA calculations. The fuel cladding failures in those events are at high temperatures and low pressures and there is adequate data to assess and approve the model for LOCA conditions. There is limited data available for transients, and the limited data for code assessment is not in the TRACG validation documentation.

TVA utilized the TRACG containment methodology for its ECCS cooling performance calculations. The methodology uses conservative biasing for both the containment and ECCS performance calculations. The methodology has been approved for containment performance calculations but it has not been approved for ECCS performance calculations and it currently does not meet the requirements of an ECCS EM as prescribed in 10 CFR 50.46. A requirement of ECCS performance calculations is to identify the limiting break locations. Using the PSAR information as currently defined, the staff identified break locations and sizes in the ICS return line and the CUW line that were not evaluated in the PSAR calculations (e.g., see RAI 15.5-2 (ML26111A161, non-proprietary; ML26111A162, proprietary), Section 15.5.3.3.2 of this report, and the response to audit question A-15.2-4 (ML26091A346)). These unanalyzed break locations and sizes may or may not be limiting breaks. In the response letter, the applicant stated that these break locations will be analyzed and presented in the FSAR and will satisfy the BWRX-300 LOCA acceptance criteria. The staff will review the assumed LOCA break locations and sizes at the OL phase to ensure an appropriate range of breaks are assumed in the analysis.

Summary of Staff Review of BWRX-300 TRACG Application

The BWRX-300 TRACG Application technical report is a description of how TRACG, combined with other codes, is applied and why it is qualified to perform BWRX-300 calculations for LOCA and non-LOCA events. The report provides an overview of the historical development and code qualification of TRACG for operating reactors, SBWRs, and ESBWRs, with references to the supporting documents. It also describes how TRACG, with links to other codes, will be used for the BWRX-300 Chapter 15 analyses for AOO, DBA, and DEC events that do not involve core damage (events that are not severe accidents). The document closely follows the approach described in the approved topical report NEDE-32906P-A, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis," but does not address how certain aspects of this previously approved topical report would apply to the BWRX-300. These aspects would need to be addressed for the BWRX-300 TRACG Application to align with the guidance in RG 1.203 and SRP Section 15.0.2. The staff notes that in PSAR Section 1.9, the applicant states their intention to comply with both RG 1.203 (see PSAR Table 1.9-20) and 15.0.2 (PSAR Table 1.9-15). Some particular differences between BWR TRACG Application and previously approved methodologies include:

- The BWRX-300 TRACG Application report identifies the analysis methods to be used for each event type, but these methods have not been approved by the staff. The report discusses previously approved methodologies that have been applied in the past to provide the technical basis for LOCA and non-LOCA analyses, but they are not used for the PSAR analyses. The BWRX-300 TRACG Application methodology uses qualitative judgement to determine margin and selects the method of analysis based on the judgement that is made. In many of the PSAR analyses, neither conservative bias nor uncertainty analysis was applied to establish a quantified basis, which is inconsistent with how TRACG has been used in previously approved methodologies. The BWRX-300 is a new reactor design that has significant differences from operating BWRs and the ESBWR in terms of the safety systems and the plant response to transients, accidents, and LOCAs. An uncertainty screening analysis would be required to make a quantitative evaluation of how much margin is available and determine what the most important sensitivities to variations in initial conditions and PIRT multipliers are for a given event.

- There are no demonstration calculations in the BWRX-300 TRACG Application showing how the methodology will be applied in practice and the BWRX-300 TRACG Application report does not prescribe how the safety analysis methodology can be followed from start to finish in a consistent manner for the BWRX-300. The DSA calculations in the PSAR are the demonstration of how TRACG for BWRX-300 is used in practice. In the PSAR, a subjective judgement about the margin is made for each calculation and that determines the category. Most of the event analyses were categorized as having large margin and use baseline calculations with nominal initial conditions.

- When applied in the PSAR safety analysis, BWRX-300 TRACG Application generally specifies that only baseline calculations with nominal initial conditions are performed. Baseline calculations with nominal initial conditions, rather than conservative calculations with initial conditions that reflect variations within the normal operating bands, are generally not appropriate for the FSAR safety analyses performed to assess risk to public health and safety.

- Calculations with some important parameters biased in a conservative direction are performed for a limited number of cases, but there is no technical basis for how the biasing parameters and ranges are selected in the BWRX-300 TRACG application report and there is not a demonstration showing that the biasing gives conservative results compared to a full uncertainty analysis.

- The TRACG fuel rod model has not been shown to calculate results equivalent to the approved PRIME model it replaces for cladding strain and pellet cladding interactions in AOO use.

- Some of the transient DBA calculations result in fuel rods heating up. There is no assessment in the TRACG documentation that validates the code in these circumstances and no demonstration that using LOCA cladding criteria is adequate for failure prediction of cladding during high-power, high-pressure transients that could have clad failure due to PCMI. GVH provided limited information during the audit that shows TRACG can give accurate heatup predictions for non-LOCA calculations under some conditions.

- The LOCA analysis is not performed with an approved LOCA evaluation model that has been shown to comply with 10 CFR 50.46. It uses the evaluation model from the containment methodology topical report.

TRACG has been used in approved BWR evaluation models for a wide range of transient and accident calculations and has a wide validation base against experimental facilities and plant data to support the use of TRACG for safety analysis as documented in the references provided in Chapter 1 of BWRX-300 TRACG Application. Part of the validation for TRACG is comparing the calculations of plant transients to the measured plant response for a variety of scenarios and TRACG generally gives accurate results for those cases. The staff also notes that the important phenomenon determining the analysis results in the BWRX-300 are the same as in the scenarios the approved methodologies have been applied to in other reactor designs. Therefore, the staff has confidence that TRACG, in the way it has been applied in the PSAR analyses, provides reasonable calculation results for the BWRX-300 PSAR safety analysis calculations in support of the CP. This makes the use of TRACG for the AOO and DBA calculations in the PSAR sufficient for the CPA demonstration of how the safety systems will function to mitigate events.

For the safety analysis for the OLA, the applicant or GVH will need to submit the TRACG methodology for staff review and approval and address the items listed above. Addressing these items will require further development of the evaluation model and some additional assessment of TRACG to have evaluation models that can be approved for the OL safety analysis. Alternatively, the approved TRACG application methodologies with additional work to support an adequate demonstration of applicability for BWRX-300 would be feasible, as was done with the use of TRACG for the ESBWR safety analysis. To ensure a more timely review at the OL phase, any unapproved methodologies based on the BWRX-300 TRACG Application report can be submitted to the staff for review in advance of an OLA.

15.11.4 Conclusion

The staff determined that the use of TRACG for the safety analyses presented in Chapter 15 of the PSAR is sufficient to support issuance of CP in accordance with 10 CFR 50.34(a). However, as noted above, the staff identified several areas where additional information is needed to complete the safety analysis at the OL stage. Consistent with 10 CFR 50.35, this further technical detail can be left for later consideration at the OL phase.