TVA Clinch River Construction Permit (CP) Application

Readiness Assessment Observations on the Draft Preliminary Safety Analysis Report (PSAR)

Chapter 3 - Design of Structures, Components, Equipment, and Systems

Section	Basis for Observation/Comment	Readiness Assessment Observations
3.1	General	See General Design Criteria (GDCs) and Principal Design Criteria (PDCs) section at end of observations table
3.3 - Wind and Tornado Loadings	10 CFR Part 50, Appendix A General Design Criteria 2, Design Bases for Protection Against Natural Phenomena	Draft PSAR Section 3.3.1.2 states that the BWRX-300 design uses ASCE 7-16. However, Section 3.3.2.2 compares the BC-TOP-3-A criteria to ASCE 7-05. A justification for not following the same code edition should be provided in the PSAR.
3.3 - Wind and Tornado Loadings	10 CFR Part 50, Appendix A, General Design Criteria 2, Design Bases for Protection Against Natural Phenomena and Standard Review Plan 3.3,2, Tornado Loadings. Standard Review Plan 3.3.2	SRP Section 3.3.2, in part E of the acceptance criteria, describes how to calculate combined tornado effects. Draft PSAR Section 3.3 does not seem to include information about the combined tornado effects and it should be included in the PSAR.
3.3 - Wind and Tornado Loadings	10 CFR Part 50, Appendix A General Design Criteria 2, Design Bases for Protection Against Natural Phenomena	Draft PSAR Section 3.3.2.1 refers to Subsection 3.8.4.4 for Seismic II and Non-Seismic Category structures. Subsection 3.8.4.4 is for the control building (CB) and states that the CB is Category II structure design per IBC. Subsection 3.8.4.4 further states that the CB is also evaluated for seismic and wind interaction with the reactor building (RB). The PSAR should clarify whether all the Seismic II and Non-Seismic Category structures share the same augmented requirements described for the CB in Section 3.8.4.4.

3.3 - Wind and Tornado Loadings	10 CFR Part 50, Appendix A, General Design Criteria 2, Design Bases for Protection Against Natural Phenomena	Draft PSAR Section 3.3.2.2 states that "the RB is an enclosed (unvented) structure and that the exposed exterior roof and walls are designed for the full pressure drop". However, additional information should be provided in the PSAR on how the pressure differences created between the interior and the exterior of the structure were considered.
3.3 - Wind and Tornado Loadings	10 CFR Part 50, Appendix A, General Design Criteria 2, Design Bases for Protection Against Natural Phenomena	Draft PSAR Section 3.3.2.3 states that the SSC that are not designed to withstand the tornado loading are arranged and designed to ensure no gross failure under the controlling wind loading. However, the applicant should include additional information that demonstrate that its failure will not result in structural damage to safety-related structures, systems, and components.
3.4 - Water Level (Flood) Design & Internal Flood Protection	10 CFR Part 50, Appendix A, General Design Criteria 2, Design Bases for Protection Against Natural Phenomena	Draft PSAR Section 3.4 referenced other draft PSAR subsections that have information required in Section 3.4, but those other subsections were not available in TVA's eRR and staff were not able to confirm if the subsections were complete. Note that two of the COL action items for the ESP SER Section 2.4 are related to establishing design basis external flood level and flood protections, which are in turn used in Chapter 3.4. The applicant should provide information in the PSAR on how it will address these two COL Action items related to flood hazard evaluation.
3.4 - Water Level (Flood) Design & Internal Flood Protection	10 CFR Part 50, Appendix A, General Design Criteria 2, Design Bases for Protection Against Natural Phenomena. Standard Review Plan 3.4.1; and Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition).	The PSAR should provide additional details of flood analysis methodology and approach to address items such items such as: 1. Describe limiting flood sources (pipe routing, tank, or other water source) 2. Location of SSCs subject to flooding. 3. Describe analysis methodology, such as "level-by-level" and "room-by-room" details of limiting/bounding source and failure considered. 4. Describe protective and mitigation features provided or credited. Design features that will be used to mitigate the effects or spread of internal flooding. 5. Describe building elevation flooding evaluated. 6. Describe features credited within building and impact on other levels or areas to establish bounding flood levels.

		7. Considers possible flow paths from non-safety related areas into areas containing safety related SSCs.
3.5 - Missile Protection	10 CFR Part 50, Appendix A, General Design Criteria 2, Design Bases for Protection Against Natural Phenomena Standard Review Plan, Sections 3.5.1.1 Internally Generated Missiles (Outside Containment) and 3.5.1.2, Internally Generated Missiles (Inside Containment)	Draft PSAR Sections 3.5.1.1 and 3.5.1.2 define internal generated missile protection. Draft PSAR Section 3.5.1.1 states, “Potential missiles that could result from failure of pressurized components are characterized as contained fluid energy missiles or stored energy (elastic missiles). These potential missiles are conservatively evaluated against the design criteria.” The draft PSAR indicates missiles analyzed include rotating equipment, high-pressure system ruptures, and missiles caused by, or as consequence of, gravitational effects (e.g., heavy load drop). PSAR Sections 3.5.1.1 and 3.5.1.2 should describe design criteria and type of missile sources evaluated, including justification for declaring missile source as credible or non-credible. For potential missiles considered, the method of SSC protection should be defined.
3.5 - Missile Protection	10 CFR Part 50, Appendix A, General Design Criteria 2, Design Bases for Protection Against Natural Phenomena Standard Review Plan (SRP) Sections: 3.5.1.1, Internally Generated Missiles (Outside Containment); 3.5.1.2, Internally Generated Missiles (Inside Containment); and 3.5.1.3, Turbine Missiles.	Draft PSAR Section 3.5.1 indicates that portions of SRP Sections 3.5.1.1, 3.5.1.2 and 3.5.1.3 are used to define parameters, but it is unclear which portions are applicable. The document contains statements, “…as defined in Nuclear Regulatory Reports (NUREG)-0800, Standard Review Plan (SRP) 3.5.1.1 through 3.5.1.3” and “…applicable portions of new NUREG-0800, SRP 3.5.1.1 through 3.5.1.3.” These SRP sections describes potential missiles and contain specific criteria to define statistical significance of an identified missile using probabilistic analysis. The PSAR should elaborate on the specific SRP guidance and method for defining statistically significant SSCs for important to safety, including possible non-safety related SSCs where failure could impact an intended safety function of the safety related SSCs.
Section 3.6 - Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping	10 CFR Part 50, Appendix A, GDC4	Section 3.6.1.1.4 of the PSAR should include clarification of what are “indirect” dynamic effects.
3.7 – Seismic Design	Technical judgment	While it is stated in Section 3.7.1.1.2 "time histories compatible to the bounding design ground motion spectra in Figure 3.7-3 are generated using the time domain spectral matching method…," the PSAR should

		reference the published documents (papers, reports, etc.) that this method is based upon.
3.7 – Seismic Design	NUREG-0800, Section 3.7.1.II.1.B, states "For generated time histories, it should be demonstrated that acceleration, velocity, and displacement are compatible and do not result in displacement's baseline drift."	Section 3.7.1.1.2 of the draft PSAR does not include the modified Time Histories (TH), acceleration, velocity, and displacement plots to demonstrate there is no displacement baseline drift.
3.7 – Seismic Design	10 CFR 50.34(a)(3)(iii), states "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety." NUREG-0800, Section 3.7.1.II.3, states "To be acceptable, the description of supporting media for each seismic Category I structure should include foundation embedment depth, depth of soil over bedrock, soil layering characteristics, design groundwater elevation, dimensions of the structural foundation, total structural height, and soil properties such as shear wave velocity, shear modulus, material damping, including strain-dependent effect, as well as Poisson's ratios, and density as a function of depth."	For the foundation embedment depth, dimensions, and total structural heights, Section 3.7.1.3 of the draft PSAR refers to Figures 3.8-1 and 3.8-9. However, the foundation embedment depth dimensions, which are required by SRP 3.7.1, is not able to be identified from these two figures. The foundation embedment depth dimensions should be identifiable in the PSAR.
3.7 – Seismic Design	Technical judgment	While Section 3.7.2.1 of the draft PSAR states: "five sets of three input motion ATHs are used as input for the SSI analyses to mitigate the uncertainty in the computed responses due to phasing of the time history frequency components.", there is no discussion of how the computed

		responses from these five sets of ATHs are combined to obtain the final design responses (e.g., member forces, stresses, ISRS etc.). The PSAR should include a discussion of how the computed responses from these five sets of ATHs are combined to obtain the final design responses.
3.7 – Seismic Design	NUREG-0800, Section 3.7.2.II.2, Natural Frequencies and Responses	In Section 3.7.3.4.2, it is not clear to the NRC staff whether sufficient modes are included to achieve high percentage (e.g., 90%) mass participation. There is no statement or discussion associated with this technical requirement. An elaboration of this statement in the PSAR would provide more clarity for NRC staff.
3.7 – Seismic Design	NUREG-0800, Section 3.7.2.II.9, Effects of Parameter Variations on Floor Response Spectra	1): While sensitivity analyses are performed to assess the effect of the variations of various analysis parameters and conditions, there is no discussion in the draft PSAR of the sensitivity results or conclusions on the selections of the sets of the analyses that are used in the final design evaluation.
3.7 – Seismic Design	NUREG-0800, Section 3.7.2.II.1.B, states "To obtain an equivalent static load for an SSC that can be represented by a simple model, a factor of 1.5 is applied to the peak spectral acceleration of the applicable ground or floor response spectrum."	In draft PSAR subsection 3.7.3.1.3, it states "Response loads are determined statically by multiplying the equipment mass by a static coefficient equal to 1.5 times the maximum spectral acceleration that corresponds to the first mode of the equipment in accordance with NUREG-0800, SRP 3.7.2, Subsection II.1.B.". This statement is not correct as the NUREG-0800 specifies the use of peak spectral acceleration as opposed to the spectral acceleration that corresponds to the first mode of the equipment. Justification of using the spectral acceleration corresponding to the first mode of equipment should be provided in the PSAR.
3.7 – Seismic Design	NUREG-0800, Section 3.7.3.II.9, Multiply Supported Equipment and Components with Distinct Inputs	1): Section 3.7.3.9 states that there are two acceptable methods for time history analysis. However, the descriptions provided for each method do not clearly differentiate between them. To improve clarity, this distinction should be elaborated on in the PSAR. Additionally, the PSAR should include criteria for selecting which method to use in the analysis. 2): The description of the response spectrum method does not appear to be related to the consideration of distinct inputs when applying response spectrum method. The requirements/acceptance criteria specified in NUREG-0800, Section 3.7.3.II.9 should be included in the PSAR.
3.7 – Seismic Design	10 CFR Part 50, Appendix S, Paragraph IV.a.4, requires that suitable instrumentation be provided	Section 3.7.4 of the draft PSAR refers to the BWRX 300 seismic instrumentation system. If there is a reference(s) to this specific instrumentation system (e.g., specific BWRX-300 design documents), the

	to promptly evaluate the seismic response of nuclear power plant features important to safety after an earthquake. Appendix S (Paragraph IV.a.3) also requires shutdown of the nuclear power plant if vibratory ground motion exceeds that of the Operating Basis Earthquake (OBE) occurs.	reference(s) should be provided since draft PSAR Section 3.7.4.2, "Location and Description of Instrumentation" does not provide a description of the proposed seismic instrumentation.
3.7 – Seismic Design	10 CFR Part 50, Appendix S (Paragraphs IV.a.3 and IV.a.4)	Draft PSAR Section 3.7.4.2 states that the location of the instrumentation follows the guidance of RG 1.12, however, no details regarding the specific locations of the instrumentation are provided. As such, the specific locations of all proposed instrumentation should be provided in the PSAR.
3.7 – Seismic Design	10 CFR Part 50, Appendix S (Paragraphs IV.a.3 and IV.a.4)	The title of draft PSAR Section 3.7.4.2.2, "Structure and Equipment Instrumentation," suggests that some instruments are located at elevations in structures as well as on equipment. No instrumentation location specifics are provided. Section B (Background) of RG 1.12, notes that "instrumentation is not located on equipment, piping, or supports since experience has shown that data obtained at these locations are obscured by the vibratory motion associated with normal plant operation." For this reason, the locations of the in-structure instrumentation should be clarified in the PSAR.
3.7 – Seismic Design	10 CFR Part 50, Appendix S (Paragraphs IV.a.3 and IV.a.4)	Draft PSAR Section 3.7.4.2.2 states that "for sensors installed in inaccessible areas, provisions for data recording and an external remote alarm indicating actuation are provided". The PSAR should indicate which sensors will be located in inaccessible areas and whether their design will allow for remote in-service testing.
3.7 – Seismic Design	10 CFR Part 50, Appendix S (Paragraphs IV.a.3 and IV.a.4)	Draft PSAR Section 3.7.4 refers directly to ANS 2.23, rather than RG 1.166. Although RG 1.166 endorses ANS 2.23, clarifications are provided in Section C of this RG. As such, the PSAR should address these clarifications.
3.7 – Seismic Design	10 CFR Part 50, Appendix S (Paragraphs IV.a.3 and IV.a.4)	Draft PSAR Section 3.7.4 refers to RG 1.12, RG 1.166, and ANSI/ANS-2.23, and states that the instrumentation and post-earthquake actions of these RGs and standards are followed. However, in many places, more specificity with respect to how these RGs and standards will be followed,

		is needed. Any differences should also be highlighted. Examples of where more specificity is needed include the following: • In PSAR Section 3.7.4.2, details regarding the seismic instrumentation type and location (i.e., Section C.1.2 of RG 1.12) are lacking. • With respect to instrumentation characteristics, PSAR Section 3.7.4.2.3 states that the instrumentation characteristics follow the guidelines in RG 1.12, Section C.4. However, only Sections C.4.3, C.4.6, and C.4.7 are mentioned. • With respect to RG 1.166 and ANSI/ANS-2.23 see comment below.
3.7 – Seismic Design	10 CFR Part 50, Appendix S (Paragraphs IV.a.3 and IV.a.4)	With respect to pre-earthquake planning, shutdown, and post-earthquake actions, Section 3.7.4.4 of the draft PSAR references ANSI/ANS-2.23 and RG 1.166. However, only Sections 6.3.1, 6.3.2, 7.1, 7.2, 8.1, and 8.2 of ANSI/ANS-2.23, as well as Appendix A of RG 1.166 are explicitly mentioned. The PSAR should clarify if other sections of this standard will be followed.
3.7 – Seismic Design	10 CFR Part 50, Appendix S (Paragraphs IV.a.3 and IV.a.4)	Section 3.7.4.4 of the draft PSAR states that the "plant is shut down if the walkdown inspections discover damage to equipment that would affect the safe operation of the plant, or if the Operating Basis Earthquake (OBE) or Safe Shutdown Earthquake (SSE) exceedance criteria listed in Sections 6.3.1 and 6.3.2 of ANSI/ANS-2.23, respectively, are met." However, this statement is not consistent with Section 6.2 of ANSI/ANS-2.23, which states that if "the OBE exceedance criterion is exceeded and/or any significant damage to safety related (SR) and non-SR systems, structures, or components (SSCs) important to safe plant operation is observed, an orderly plant shutdown is required for U.S. plants." As such, the PSAR should clarify the shutdown criteria.
3.8 – Design of Category I Structures	ASME Section III, Table CC-3230-1	According to ASME Section III, Table CC-3230- 1, Load case Ha - internal flooding is included in load combinations for the abnormal/severe environmental category. However, the Ha load case is not included in Table-3.8-1 for SCCV. The table in the PSAR should indicate whether internal flooding load applies to the SCCV containment walls design.
3.8 – Design of Category I Structures	NUREG-0800, Section 3.8.2.II.4.C, Computer Programs	The draft PSAR does not identify the computer program(s) that is/are used to perform the FE analyses of Class MC steel components.

3.8 – Design of Category I Structures	NUREG-0800, Section 3.8.3.I.7, Testing and Inservice Surveillance Programs.	Although Section 3.8.3.7 stated that a formal program of testing and surveillance of containment internal structures is not required, the PSAR should address accommodations for access to critical areas and penetrations that require inspections.
3.8 – Design of Category I Structures	Technical Judgment	Based on Figure 3.8-2, the reactor building foundation and the containment foundation appears to be two structural elements with different materials. Since the containment foundation (SCC member) is integral with the RB (reinforced concrete) mat foundation, appropriate reinforcement details need to be considered to provide a monolithic connection. The draft PSAR should evaluate this joint for forces from all possible loading combinations and conditions.
3.8 – Design of Category I Structures	10 CFR Part 50, Appendix A GDC 2	Subsection 3.8.4.4.1 states that the Control Building that houses the main control room and electrical control and instrumentation equipment is a Category 2 structure. RG 1.29 - Seismic Design Classification for Nuclear Power Plans, Part C, Subsection 1.d, bullet 5 lists "the control room, including its associated equipment and all equipment needed to maintain the control room within safe habitability limits for personnel and safe environmental limits for vital equipment." as examples of Category 1 structure. The PSAR should clarify whether the Control Building discussed in Section 3.8.4.4.1 is appropriately categorized as seismic Category 2 or 1.
3.9 - Mechanical Systems and Components	10 CFR 50.55a RG 1.100	1. ASME QME-1-2017 is the latest version of ASME QME-1 endorsed in RG 1.100 at this time. Additionally, the 2021 Edition is the current version of the ASME BPV Code incorporated by reference in 10 CFR 50.55a. TVA references the 2021 Edition of the ASME BPV Code in Section 3.9 of the PSAR. The NRC will be updating 10 CFR 50.55a and RG 1.100 for more recent editions of the ASME BPV Code and ASME QME-1 Standard in the future. 2. Section 3.9 should include a table identifying the active valves and a table identifying the in-service testing activities for applicable pumps and valves.
3.9 - Mechanical Systems and Components	10 CFR Part 50, Appendix A RG 1.100.	1. Section 3.9.1.4 should include references to the functional qualification of valves per ASME QME-1 as accepted in RG 1.100.

	10 CFR 50.55a	2. The PSAR should clarify whether there are any essential safety-related systems and piping that need to be designed to normal stress limits for plant faulted condition. 3. The PSAR should clarify whether the containment penetration sleeve anchors are designed to combined loads from inside containment (CTMT) and outside CTMT loads. Also, applies to 3.12.1.
3.9 - Mechanical Systems and Components	10 CFR 50.55a	Section 3.9.2.1, “Piping Vibration, Thermal Expansion and Dynamic Effects” should reference the relevant standards. Typically, applicants reference ASME Operation and Maintenance Code Part 3, “Vibration Testing of Piping Systems” and Part 7, “Thermal Expansion Testing of Nuclear Power Plant Piping Systems.”
3.9 - Mechanical Systems and Components	10 CFR Part 50, Appendix A	Subsection 3.12.3.13 states "Combination of inertia and seismic anchor motion effects are discussed in Subsection 3.9.3.6.1. However, Subsection 3.9.3.6.1 does not discuss the combination of inertial and seismic anchor motion effects and this should be reconciled in the PSAR.
3.9 - Mechanical Systems and Components	10 CFR 50.55a	FIV loads associated with transients are evaluated as part of alternative service level. The PSAR should clarify what this alternative service level is.
3.9 - Mechanical Systems and Components	10 CFR Part 50, Appendix A RG 1.100	Section 3.9.3.12 in the PSAR should: (1) clarify the second paragraph regarding use of parentheses. (2) specify that ASME QME-1 will be used as accepted by RG 1.100 for the applicable valves (3) indicate the specific valves that will be qualified using QME-1 per RG 1.100 and the valves that will be qualified by a different method (4) specify the approach for the qualification of passive valves to meet the NRC regulations.
3.9 - Mechanical Systems and Components	10 CFR Part 50, Appendix A RG 1.100	Section 3.9.3.14.2 in the PSAR should specify the use of ASME QME-1 as accepted in RG 1.100 for qualification testing. For example, this section of the draft PSAR includes outdated qualification language that only refers to time limits and differential pressure in demonstrating the functional capability of valves.

3.9 - Mechanical Systems and Components	10 CFR Part 50, Appendix A RG 1.100	Section 3.9.3.14.3 in the PSAR should specify the use of ASME QME-1 as accepted in RG 1.100 for qualification testing because the draft PSAR language appears out of date.
3.9 - Mechanical Systems and Components	NUREG-0800, SRP 3.9.4.	Section 3.9.4.4, "Control Rod Drive Performance Assurance Program," primarily discusses the integrity function of the reactor coolant pressure boundary (RCPB) components of the control rod drive mechanism (CRDM) but has insufficient discussion of the plans for an operability assurance program, as described in SRP 3.9.4, Control Rod Drive Systems.
3.9 - Mechanical Systems and Components	10 CFR 50.34	Section 3.9.5, "Reactor Pressure Vessel Internals," or other sections of the PSAR should provide more figures to allow the staff to understand the basic design of the reactor internals.
3.9 - Mechanical Systems and Components	10 CFR 50.55a	Section 3.9.6 of the PSAR should: (1) specify that the ASME OM Code must be implemented as incorporated by reference in 10 CFR 50.55a (2) indicate the pumps, valves, and snubbers that will undergo in-service testing in accordance with 10 CFR 50.55a.
3.10 - Seismic and Dynamic Qualification of Mechanical and Electrical	IEC/IEEE 60980-344 IEEE 382	Section 3.10.3.3, "line-mounted equipment," refers to IEC/IEEE 60980-344 and IEEE 382. Line-mounted equipment may also include active mechanical equipment that may also be subject to ASME QME-1 qualification. Consider whether discussion of ASME QME-1 may be appropriate in this section of the PSAR.
3.10 - Seismic and Dynamic Qualification of Mechanical and Electrical	10 CFR Part 50, Appendix A RG 1.100, Revision 4	Section 3.10.1.1.1, "Qualification by Actual Seismic Experience," states that IEC/IEEE 60980-344, Annex A provides experience based seismic qualification methodology and is utilized as appropriate. The NRC stated the staff positions in RG 1.100, Revision 4, regarding the necessary information for the review of the use of earthquake experience for the seismic qualification of electrical equipment apply. The NRC staff plans to update RG 1.100 to address the most recent edition of IEEE 344.
3.10 - Seismic and Dynamic Qualification of Mechanical and Electrical	10 CFR Part 50, Appendix A	Section 3.10.2.2.1, "Interface Requirements," states that "The effects of intervening structures or components that serve as interfaces between the equipment to be qualified and that are supplied by others are not qualified as part of the program." Section 3.10.2.2.4, "Mounting of Test Specimen" states that "The test specimen is mounted to the test table so

		that inservice mounting, including interfaces, is simulated." It is not clear that these statements are consistent. The NRC staff notes that IEC/IEEE 344-60980-2020, "Nuclear Facilities – Equipment Important to Safety – Seismic Qualification," Section 9.1.2, "Mounting," states that the effect of electrical connections, conduit, sensing lines, and any other interfaces shall be considered and included in the setup up to the first interface support unless otherwise justified.
3.11 - Environmental Qualification of Mechanical and Electrical Equipment	10 CFR Part 50, Appendix A, GDC 4 NUREG-0800, SRP 3.11 RG 1.89	Section 3.11 discusses mild and harsh environments. The description of a harsh environment appears to address the definition in 10 CFR 50.49. However, it does not address the potential of equipment degradation due to long term radiation exposure regardless of the change in the environment during accident conditions. GDC 4 specifies that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The PSAR should address the potential degradation of equipment due to cumulative radiation exposure, even if the accident conditions are not significantly different than normal operating conditions. For example, SRP 3.11 and RG 1.89 provides guidance on the total integrated doses which the staff has previously considered acceptable thresholds for determining if an environment is mild or harsh for the purposes of determining if the equipment may be degraded from long term radiation exposure.
3.11 - Environmental Qualification of Mechanical and Electrical Equipment	10 CFR Part 50, Appendix A, GDC 4	It is unclear what assumptions are made for determining the total integrated doses that equipment is expected to receive and what total integrated doses equipment are expected to be exposed to. The assumptions made should be clarified in the PSAR.
3.11 - Environmental Qualification of Mechanical and Electrical Equipment	ASME QME-1	The topic of mechanical equipment qualification is discussed in 3.11.2.5.1, but the proposed methodology used is not clear. The PSAR should clarify if ASME QME-1 is the intended methodology, and it should be specified if nonmandatory appendices (such as QR-B for non-metallics) will be used.
3.12- ASME BPV Code Class 1, 2, and 3 Piping Systems and Associated Support Design	10 CFR 50.55a	1. The PSAR should clarify the decoupling criteria used for Piping analysis in terms of diameter ratio, section modulus ratio, or moment of inertia ratio. 2. The PSAR should clarify whether seismic loading from OBE is a part of Service Level B evaluations.

3.13 - Threaded Fasteners—ASME BPV Code Class 1, 2, and 3	NUREG-0800, SRP Section 3.13.	It would aid clarity and facilitate the review if the PSAR summarized the provided ASME Code citations in a table, organized such as the Table 3.13-1 of NUREG 0800, SRP 3.13. The textual citation of the respective ASME Code requirements is somewhat open to interpretation. Providing descriptions of any novel bolting features that exist in the design, such as permanently affixed tooling fixturing, would be helpful in review of this section.

Chapter 4 - Reactor

Section	Basis for Observation/Comment	Readiness Assessment Observations
4.0 – Reactor Design	10 CFR 50.46; 10 CFR Part 50, Appendix A, GDC 10 and GDC 13; 10 CFR 50.61; 10 CFR 50.61a; 10 CFR Part 50, Appendix H	The nuclear instrumentation description in the PSAR should include the detector types at the different power ranges.
4.2 – Fuel System Design	Editorial and Clarity	The draft PSAR Section 4.2.4 statement "Additionally, ICS [isolation condenser system] protects against overpressure protection that further maintains coolable geometry that meets criterion of 10 CFR 50.46(b)(4)" is unclear. Consider removing one of the mentions of protect/protection.
4.2 – Fuel System Design	10 CR 50.46; 10 CFR Part 50, Appendix A, GDC 10 and GDC 27	The following information should be provided as part of reports incorporated by reference into the CPA, or directly in Chapter 4 of the PSAR: • Justifications regarding the applicability of the fuel performance evaluation codes cited. • A detailed fuel performance evaluation. • A demonstration of compliance with SAFDLs, such as rod internal pressure at the end of fuel assembly life. • A cladding fatigue analysis for the BWRX-300 core.

		• Applicability of the referenced control rod design evaluation to the BWRX-300 design. Note: This information may be contained in References 4.2-9 and 4.2-10, which were not available for staff review as part of the readiness assessment.
4.3 – Design Basis	10 CFR 50.34; 10 CFR 50.46; 10 CFR Part 50, Appendix A, GDC 10	Nuclear computer code benchmarking information for the BWRX-300 design should be provided in the PSAR or in reports incorporated by reference into the CPA. Reference 4.3-2 for TGBLA06/PANAC11 applicability to BWRX-300 may contain this information, however this was not made available for the readiness assessment. The benchmarking analyses, or applicability of benchmarking to the BWRX-300 design, are needed for the NRC-approved nuclear analysis computer codes to determine the biases and uncertainties of the codes when applied to analyzing the BWRX-300 reactor because of the differences in designs and working principle.
4.3 – Design Basis	10 CFR 50.34; 10 CFR 50.46; 10 CFR Part 50, Appendix A GDC 10	The following information should be provided (either as part of a document incorporated by reference into the CPA, or as part of Chapter 4 of the PSAR): • Fuel bundle and core enrichments and burnable poison loadings for the initial cycle or the equilibrium cycle, at a minimum. • Integrated and differential control rod worth information. • Key nuclear design parameters such as the effective delay neutron fraction βeff, average prompt neutron lifetime Λ, Doppler reactivity coefficients (max and min). • Key core performance parameters at BOC, middle of cycle (MOC), and end of cycle (EOC) as a function of power level based on a nominal cycle depletion. Note: This information may be provided in reference 4.3-1 to NEDC-34044P, “BWRX-300 GNF2 Equilibrium 12-month cycle nuclear design report,” Revision 0, November 2023, but that report was not included in the readiness assessment.
4.4 – Thermal and Hydraulic Design	10 CFR 50.34	The draft PSAR discusses the methodologies the applicant intends to use for thermal and hydraulic design of the BWRX-300 reactor. However, there is no information on the calculations for the thermal and hydraulic design. Information for the thermal analyses should be provided and

		include descriptions of the models, the input parameters, the values used in the calculations and the results of the calculations. Some of this information may be included in References 4B-1 and 4.4-3, however these were not available for staff review for the readiness assessment.
4.5 – Reactor Materials	NUREG-0800, SRP Section 4.5.1 and 4.5.2. ASME Code Sections II and III	Recent international experience corroborates that modern low sulfur 3XX steels may be more susceptible to stress corrosion cracking than generally acknowledged. As noted in SRP 4.5.2, Section 1, under “Materials,” the adequacy and suitability of the materials specified are reviewed for, among several aspects, stress corrosion resistance. Discussion in the PSAR of how the potential for modern low sulfur 3XX steels’ relatively larger susceptibility to stress corrosion cracking (relative to historical 3XX steel supplies) will assist in addressing this review area. NOTE: A big part of review in 4.5.1 and 4.5.2 is material suitability for the operating environment. The recent stress corrosion cracking in low sulfur steels in France is not an isolated event and mentioning it as an observation/comment flows from the review guidance in SRP 4.5. The ASME Code was developed when steel impurity control was less stringent, so sulfur content is often not mentioned in ASME Code Section II, especially for older specs. 3XX steels are in both the CRD and internals sections.
4.5 – Reactor Materials	NUREG-0800, SRP Section 4.5.1 and 4.5.2	Draft PSAR Sections 4.5.1 and 4.5.3 do not include diagrams of the reactor vessel internal components and core supports or reference diagrams that may exist elsewhere in the application. These diagrams will be important in enabling the staff to make findings for Sections 4.5.1 and 4.5.3. and should be included in the PSAR so staff can better understand the general shape, design, and operating environment of all the subject components, especially those listed in Tables 4.5-1 and 4.5-2. NOTE: Such diagrams need not be dimensionally precise or fully accurate but are necessary for staff to confirm their understanding of the subject components, interfaces between these components, and operating environment of subject components in order to make findings under the general design criteria and as described in SRP 4.5.1 and SRP 4.5.2. Sole reliance on textual descriptions has not proven sufficient in previous reviews for these sections due to the complexity of the subject components.
4.5 – Reactor Materials	10 CFR Part 50.55a	Draft PSAR section 4.5.3.1 does not explicitly cite ASME Code Section II but indicates that materials listed in Table 4.5-1 are merely

	NUREG-0800, SRP Section 4.5.1 and 4.5.2. ASME Code Sections II and III	“representative.” This appears inconsistent with SRP 4.5.2 which relies on the use of ASME Code citations to underpin NRC review (and for incorporation by reference of ASME Code requirements in 10 CFR 50.55a). There description given is not sufficient for NRC staff to draw conclusions regarding this in Section 4.5.3.1 and the PSAR should be revised accordingly.
4.5 – Reactor Materials	RG 1.171 ASME Code Sections IX	Section 4.5.3.2 indicates that welder and welding procedures are qualified according to ASME Code, Section IX, in conformance with RG 1.71. It is unclear whether welding procedures are qualified according to ASME Code, Section IX, when used for areas without limited accessibility (the topic of RG 1.71). The PSAR should clarify the extent of conformance with ASME Code, Section IX.
4.5 – Reactor Materials	10 CFR Part 50, Appendix A, GDC 1 10 CFR Part 50, Appendix A	Section 4.5.3.2 is generally lacking in detail and specificity. High-level statements of conformance with 10 CFR Part 50, Appendix A and 10 CFR Part 50, Appendix B leave little basis for staff to make findings.
4.5 – Reactor Materials	ASME Code Sections III, sub-articles NB-2500 and NB-5000	Section 4.5.3.3 cites ASME Code, Section III, Sub-articles NB-2500 and NB-5000. It is unclear which components falling under Section 4.5.3 would be subject to the requirements of NB sub-articles.
4.5 – Reactor Materials	RG 1.44	Section 4.5.3.4 makes no mention of any testing for sensitization (e.g. use of RG 1.44). It is unclear whether this is due to an intent to select materials for which testing is not necessary. The PSAR should clarify specifically how material selection and control precludes the need for testing; include testing for sensitization; or otherwise providing a complete description of how sensitization will be avoided and the lack-thereof verified.
4.5 – Reactor Materials	NUREG-0800, SRP Section 4.5.1 and 4.5.2.	Table 4.5-1 only provides SFA numbers for weld materials, and Table 4.5-2 has no weld materials listed. This level of detail does not support review under SRP 4.5.1 or SRP 4.5.2. The PSAR should include a description of which welding specifications will be used, and whether there are or are not welds in the control rod drive components and materials.
4.6 – Design of Reactivity Control Systems	10 CFR Part 50, Appendix A, GDC 26 10 CFR 50.34	Section 4.6.4 in the draft PSAR states “An exemption is being developed demonstrating an alternative method for conformance to GDC 26 and will be provided later.” The staff agrees that the development of a PDC in lieu

		of this GDC would be needed, and the staff will review the justification for the exemption as part of the CP application review. See General Design Criteria (GDCs) and Principal Design Criteria (PDCs) Section at the end of this table for specific observations associated with GDC/PDC 26 and 27.

Chapter 5 – Reactor Coolant System and Connected Systems

Section	**Basis for Observation/Comment**	**Readiness Assessment Observations**
5.2 - Reactor Coolant Pressure Boundary Integrity	RGs 1.84, 1.147 and 1.192	Section 5.2.1.2 The draft PSAR states that Table 5.2-1, "Applicable Code Cases for ASME BPVC Components," lists the specific Code Cases endorsed in RG 1.84 that are applied to BWRX-300 ASME BPVC, Division 1, Section III components. A statement should be added to the PSAR that the identified Code Cases listed are subject to the applicable conditions specified in Table 2 of RG 1.84. Statements should also be added for RG 1.147 and RG 1.192 which state that Code Cases used and listed in Table 2 of RG 1.147 and RG 1.192 are subject to the applicable conditions established in the RG
5.2 - Reactor Coolant Pressure Boundary Integrity	Important practice for ensuring integrity and performance of a welds (lessons learned from previous design reviews)	Section 5.2.3.3.2 The PSAR should include a paragraph on welding of ferritic steels and identifying and describing post weld heat treatments and their time/temperature.
5.2 - Reactor Coolant Pressure Boundary Integrity	10 CFR 50.55a	Section 5.2.1.1, "Compliance with 10 CFR 50.55a," is within Chapter 5, entitled "Reactor Coolant System and Connected Systems." However, SRP 5.2.1.1 expects this section demonstrate compliance with 10 CFR 50.55a for the entire design, not just the reactor coolant system. The application should discuss the entirety of SSCs subject to 10 CFR 50.55a.
5.2 - Reactor Coolant Pressure Boundary Integrity	10 CFR 50.55a	Table 5.2-1 lists Applicable Code Cases for ASME BPVC Components. While some of these Code Cases are listed as acceptable for use without condition in RG 1.84, others are conditionally acceptable (or not yet approved by ASME). The application should address any conditions on

		use of the Code Cases in RG 1.84 and justify use of the Code Case not yet approved by ASME.
5.2 - Reactor Coolant Pressure Boundary Integrity	10 CFR 50.34, GDC 31	Section 5.2.2 Missing calculation to demonstrate that the reactor coolant pressure boundary will be protected with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. Section 5.2.2 points to Section 6.3.3 for information related to overpressure protection. However, the information in Section 6.3 of the draft PSAR is more on the system design goals rather than demonstration of compliance with the relevant regulations.
5.3 – Reactor Vessel	10 CFR 50.34, 10 CFR 50.55a, 10 CFR 50.60, 10 CFR Part 50, Appendix H	Section 5.3 1. Missing neutron fluence calculation. The draft PSAR references Section 4.1.6.1 for the neutron fluence calculation but this section does not present the result. 2. Missing information on code validation for the BWRX-300 RPV fluence calculation.
5.3 – Reactor Vessel	GDCs 1, 4 and 14	Section 5.3.1.4 References are made in several sections, including 5.3.1.4, to Regulatory Guide 1.44, “Control of the Use of Sensitized Stainless Steel.” This regulatory guide provides guidance regarding testing for sensitization. It would be helpful for reviewers to have TVA specify in the PSAR which testing standard will be used for testing for sensitization as reference to a regulatory guide does not provide a clear method to support an NRC finding.
5.3 – Reactor Vessel	10 CFR 50.60, 10 CFR Part 50, Appendix H	Section 5.3.1.6.1 An exemption is requested to use ASTM E185-21. The lead factors permitted by ASTM E185-21 allow for up to a 5x lead factor. Based on the information provided, the staff cannot link the proposed fluence targets in Section 5.3.1.6.1 to operational calendar years. A description of the maximum planned lead factor, or a note that such will be provided in the FSAR, would help reviewers reach an NRC finding.

5.4 – Reactor Coolant System Component and Subsystem Design	10 CFR 50.34, GDC 33, GDC 34, GDC 13	Section 5.4 1. Missing information on makeup water system. GDC 33 requires that a reactor design to include a reactor coolant makeup system for protection against small breaks in the RCS. Although the PSAR states that the control rod drive (CRD) provides a source of inventory makeup in Modes 2 to 6, it is unclear whether the CRD system can fulfill the regulatory requirements of GDC 33, since it is designated as non-safety-related, or if an exemption from GDC 33 is being sought. 2. Missing bases for excluding test requirement for isolation condenser. 3. Missing test requirement for shutdown cooling (SDC) system. 4. Missing information on requirements for compliance with GDC 13. PSAR Section 5.4.6.5 discusses instrumentation requirements. However, it does not specify how GDC 13 will be satisfied.
5.4 – Reactor Coolant System Component and Subsystem Design	SECY-94-084	Section 5.4.7 See observation in Table 15.6.1regarding missing information pertinent to addressing technical issues related to regulatory treatment of non-safety system (RTNSS) in SECY-94-084 requirements.
5.4 – Reactor Coolant System Component and Subsystem Design	SECY-90-016 and SECY-93-087	Sections 5.4.7.3 (and 3.12 and 3.18) Insufficient information on compliance with the ISLOCA requirements for all systems and subsystems (including the SDC system) connected to the RCS required in SECY-90-016 and SECY-93-087.
5.4 – Reactor Coolant System Component and Subsystem Design	10 CFR 50.34	Section 5.4.5.2, "System Description," subsection "Main Steam Subsystem" states: "Each MSL is equipped with an MSRIV and an MSCIV." However, Figure 5.4-5, “Main Steam System” shows two reactor isolation valves for each main steam line. It appears that the statement should be "Each MSL is equipped with two MSRIVs and an MSCIV".

Chapter 6 – Engineered Safety Features

Section	Basis for Observation/Comment	Readiness Assessment Observations

6.2 – Containment System	10 CFR Part 50, Appendix A, GDC 16, GDC 50, GDC 38	Section 6.2.2 As illustrated in Figure 6.2-2, the draft PSAR presents a new PCCS design that consists of three independent trains each with a passive containment cooling pipe array (PCCPA) of once-through condensing pipes vertically connected to an intake header at the bottom and a return header at the top for the reactor cavity pool subcooled water circulation. The new PCCS design and piping structure presented in the draft PSAR is different from the concentric-pipe PCCS design configuration with individual containment penetrations that was reviewed and modeled in draft PSAR Reference 6.2-2 for the demonstration of the applicability of the BWRX-300 design containment evaluation methodology (CEM). Therefore, as required by the Limitation and Condition#4 (L&C#4) documented in the Reference 6.2-2 SER, the applicability of the method and the modeling approach must be reviewed by the NRC for the PCCS design in the CP application and found to be acceptable for the BWRX-300 licensing-basis analyses. The PSAR should demonstrate compliance with L&C#4.
6.2 – Containment System	10 CFR Part 50, Appendix A, GDC 16, GDC 50, GDC 38	Sections 6.2.1 and 6.2.2 Draft PSAR Reference 6.2-2 is an approved TR that identifies two water pools in the BWRX-300 design: (1) reactor cavity pool, and (2) isolation condenser system (ICS) pool. The reactor cavity pool that is used for the passive containment cooling system (PCCS) heat removal during design.basis events (DBEs) is located above the containment head. The ICS pool is located circumferentially around the reactor cavity pool with three IC condensers to reject the decay heat. Reference 6.2-2 presented that the heat is passively rejected from the containment to the reactor cavity pool by natural circulation of the subcooled water from the reactor cavity pool through the containment using the PCCS water piping. The Reference 6.2-2 methodology clearly isolated the boundaries between the reactor cavity pool and the ICS pool, as illustrated in Figure 2-2 of Reference 6.2-2 that shows the three isolation condenser (IC) trains placed in a peripheral ICS pool located around the reactor cavity pool. However, the draft PSAR refers to an “equipment pool” as on Page 6-14, where it states, “Passive heat rejection to the equipment pool and reactor cavity pool which forms the ultimate heat sink for the PCCS.” The draft PSAR Figure 6.2-2 also shows the PCCS pipings to be rather connected to the “equipment pool” circumferentially located around the reactor cavity pool region where one would expect the ICS pool. As the term

		“equipment” pool did not exist in Reference 6.2-2, the staff assumes it could either be the ICS pool or a “newly introduced third” pool in between. As this deviation has been repeated several times in the draft PSAR, the staff would point that per Reference 6.2-2, the PCCS rejects heat only to the reactor cavity pool and not to the ICS pool that is rather designed for the decay heat rejection using ICS condensers. The GOTHIC model for the BWRX-300 containment response that the staff reviewed and approved in Reference 6.2-2 connected the PCCS piping only to the reactor cavity pool dimensions and water capacity, while the ICS pool was included in a separate TRACG model developed for the reactor systems. The PSAR should clarify the role and contribution of the newly introduced “equipment pool” vis-s-vis the approved CE methodology. The Reference 6.2-2 analysis methodology is only approved for a design that thermally isolates the reactor cavity pool and the ICS pool.
6.2 – Containment System	10 CFR Part 50, Appendix A, GDC 16, GDC 50, GDC 38	Sections 6.2.1 and 6.2.2 The draft PSAR makes no reference to meeting the conditions and limitations that were formulated by the NRC staff in the respective SERs of References 6.2-2 and 6.2-3 as a part of the LTRs approval. Another condition and limitation relevant to Sections 6.2.1 and 6.2.2 is L&C#3 in the Reference 6.2-2 SER. Per L&C#3, the use of the approved 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. So, the PSAR referencing this LTR needs to demonstrate that either 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.
6.2 – Containment System	10 CFR Part 50, Appendix A, GDC 16, GDC 50	Section 6.2.1 References 6.2-2 and 6.2-3 state that the BWRX-300 containment subcompartments include the volume below the RPV, the space between the RPV and the biological shield, and the containment head area above the refueling bellows. Draft PSAR Page 6-7 states “Pipe breaks outside the subcompartments may result in transient loads on the subcompartment boundaries. However, because the pipe breaks occur in a relatively large volume and subcompartments have large openings, these loads are not large in magnitude. The pressure differential across subcompartment boundaries for the most limiting case is found to be

		small. The design of the subcompartment includes the differential pressures resulting from pipe breaks outside the subcompartments." The staff reviewed similar information submitted for the case demonstrated in Reference 6.2-2 and approved the use of GOTHIC code and the methodology for subcompartment analysis. However, the draft PSAR statement about the pressure differentials across the subcompartment walls being small should be justified with an analysis in the CP application for the containment design under the postulated DBAs caused by line breaks either inside or outside the subcompartments.
6.2 – Containment System	10 CFR Part 50, Appendix A, GDC 16, GDC 50, GDC 38	Sections 6.2.1 and 6.2.2 Draft PSAR Section 6.2.2 on Page 6-16 states that "The PCCS has sufficient capacity to reduce the pressure and temperature in the containment, minimizing leakage following an accident and maintaining pressure and temperature below the design limits in isolation events. Chapter 15 provides the planned response to design basis events." The staff would emphasize that the containment safety analyses for containment functional design and heat removal systems should be submitted in PSAR Chapter 6 and not in PSAR Chapter 15. Containment safety under design basis accidents will be reviewed under the NRC regulations, GDC 16, GDC 50, and GDC 38, and acceptance criteria that are documented in SRP Sections 6.2.1and 6.2.2.
6.2 – Containment System	10 CFR Part 50, Appendix A, GDC 16, GDC 50, GDC 38; Appendix K; 10 CFR 50.46	As documented in the Reference 6.2-3 SER, GEH stated that the mass and energy release rates used in the BWRX-300 containment analyses will be calculated accounting for all applicable sources of energy required for consideration in 10 CFR Part 50, Appendix K. GEH did describe these applicable energy sources, correlations, and conservative biases in Reference 6.2-2. Appendix K to 10 CFR Part 50 also specifies analysis requirements for 10 CFR 50.46 acceptance criteria for the emergency core cooling system (ECCS) for light water reactor compliance. However, Appendix K is missing from the applicable regulations identified in draft PSAR Section 6.1.3. The PSAR for the CP application should include Appendix K.
6.2 – Containment System	GDC 54	Section 6.2.4 RG 1.70, Revision 3, Section 6.2.4.2 states the following: "Provide a table of design information regarding the containment isolation provisions for fluid system lines and fluid instrument lines penetrating the containment."

		This section lists 22 items to be provided on the design information. A table providing this information was not available in the draft PSAR. RG 1.70, Revision 3, Section 6.2.4.2 states the following: “Discuss the provisions for detecting leakage from a remote manually controlled system (such as an engineered-safety-feature system) for the purpose of determining when to isolate the affected system or system train.” This information was not available in the draft PSAR.
6.2 – Containment System	GDC 51	Section 6.2.7 Section 6.2.7 provides no discussion concerning fracture prevention of the containment pressure boundary, rather reference is made to other sections of the application. These referenced sections do not appear to explicitly discuss fracture prevention. An explicit discussion of fracture prevention, such as citation of the applicable and relevant ASME Code requirements for design and testing of material, in the PSAR would assist the staff in making a finding on Section 6.2.7.
6.3 – Emergency Core Cooling System	Appendices A and B to 10 CFR Part 50	Section 6.3 provides insufficient information addressing the issues related to detection, prevention and elimination of gas accumulated in the safety-related and non-safety related systems specified in GL-2008-01 and ISG of DC/COL-ISG-019 (ADAMS Accession No. ML111170302). While this ISG was written for 10 CFR Part 52 applications, it remains technically applicable to construction permits.
6.3 – Emergency Core Cooling System	10 CFR 50.35, Appendices A and B to 10 CFR Part 50	Draft PSAR Figure 6.3-4 for the hydrogen recombiner is noted as "conceptual". The PSAR is required to provide the information for a "preliminary" design, versus a "conceptual" design because NRC staff cannot make its safety determination based on a "conceptual" design. If the applicant is planning on a research and development (R&D) program for this safety feature (these phrases are quoted from 10 CFR 35), the applicant needs to follow RG 1.70 (see Section 1.5) guidance for providing the information the NRC staff needs in the PSAR describing the requirements for further technical information.
6.3 – Emergency Core Cooling System	10 CFR 50.46 and 10 CFR Part 50, Appendix A	See LOCA ECCS related observation in 15.5 - Deterministic Safety Analyses.
6.3 – Emergency Core Cooling System	10 CFR 50.34, GDC 31, GDC 54	Section 6.3.3

		Missing information on compliance with regulatory requirements of GDC 31 and GDC 54. The PSAR should list GDC 31 and GDC 54 in this section or refer to the GDCs in Section 3.1.
6.3 – Emergency Core Cooling System	10 CFR 50.34(f)	Section 6.3.3 PSAR references Enclosure 4 for requests of exemptions to 10 CFR 50.34(f), but the referenced document is not posted in the eRR and the list of preliminary exemptions to the regulations in the eRR do not include 10 CFR 50.34(f)
6.3 – Emergency Core Cooling System	10 CFR 50.34(f)	Section 6.3.3.1.2 This section is not clear whether all pertinent TMI items will be addressed and if justification will be provided for those items not deemed technically relevant.
6.3 – Emergency Core Cooling System	GDCs 17 and 27	Section 6.3.3 Missing information for supporting the request for exemptions from GDCs (including GDCs 17 and 27) in the applicant's March 7, 2024, letter posted in the eRR.
6.3 – Emergency Core Cooling System	10 CFR 50.34, GDC 31, GDC 33	Section 6.3.3 Missing demonstration for compliance with GDC 33. Section 6.3.3.1.1 states: "GDC 33 - Reactor Coolant Makeup: The ICS supports the design concept of preserving vessel inventory to cool the core. The RIVs are designed to close in a manner that preserves reactor coolant for breaks in piping that could cause an inventory loss large enough to prevent the ICS from cooling the core for 72 hours without operator action." This statement appears to indicate that makeup is not provided for the design. Therefore, the section is missing a description of an exemption from GDC 33 and the basis for such an exemption. The staff notes that the response to RAI 9732 on NEDC-33910P detailed criteria and a potential basis to support an exemption request with a proposed PDC 33 and it is not clear if this remains the planned approach or not, since NEDC-33910P remains referenced in 6.3, yet no exemption request or PDC is described either here or in the subsequently added Section 3.1.4.4.
6.4 – Control Room Habitability	GDC 19	SRP 6.4, Revision 3, Acceptance Criteria 3(B) states the following: “Systems having pressurization rates of less than 0.5 and equal to or greater than 0.25 volume changes per hour should have identical testing

		requirements as indicated in acceptance criteria 1 above. In addition, at the construction permit (CP), combined license, or standard design certification stage, an analysis should be provided (based on the planned leaktight design features) that ensures the feasibility of maintaining the tested differential pressure with the design makeup airflow rate." No information is available in the draft PSAR.
6.6 – Preservice and Inservice Inspection of Class 2 and Class 3 Components	10 CFR50.55a(g)(4)	Section 6.6.3.2 Under "Alternative Examination Techniques," the draft PSAR states that as provided in ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given component in this section, provided they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc. which may result in improvements in examination reliability and reductions in personnel exposure. A statement should be added to the PSAR which states that the newly developed techniques shall be approved by the NRC and if necessary, conditions provided by the NRC staff shall be incorporated for use.

Chapter 7 - Instrumentation and Controls

Section	**Basis for Observation/Comment**	**Readiness Assessment Observations**
7.1 - Instrumentation and Controls	10 CFR Part 50, Appendix A, General Design Criteria	Note that the acceptability of the Instrumentation and Control (I&C) systems for BWRX-300 is contingent upon acceptability of NEDC-33934P BWRX-300 Safety Strategy Topical Report currently under review by NRC staff.
7.3 – Distributed Control and Information System Functions	10 CFR 50.36, "Technical Specifications," and 10 CFR 50.55a(h) that incorporates by reference IEEE Standard 603-1991, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations"	10 CFR 50.36(c)(1)(ii)(A) states that limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions. This clause requires that where a limiting safety system setting (LSSS) is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective action will correct the abnormal situation before a safety limit is exceeded. Clause 6.8 of IEEE 603 requires a setpoint methodology. Typically, this methodology complies with RG 1.105 r/4, which endorses

		ANSI/ISA 67.04.01 – 2018. The starting point for this calculation requires an analytic limit (AL), which is provided in the safety analysis of the application to protect the safety limits (SL). Typically, tables are provided in Chapter 7 that document all safety functions protecting SLs, thus requiring a setpoint calculation. This table includes sensor range, nominal setpoint (SP), and AL. Some of this information may not be available in the CPA until the design is complete.

Chapter 8 – Electric Power

Section	Basis for Observation/Comment	Readiness Assessment Observations
8 – Electric Power	10 CFR Part 50, Appendix A, GDC 1, 2, 3, 4, 5, 17, 18	Chapter 8 shows that there are three Safety Categories of Functions/Systems/Equipment - SC1, SC2, and SC3. The PSAR should provide a list of electrical equipment by safety category - function/system - and the associated specific qualifications for each category with justification. Also, the PSAR should specify which NRC regulations will apply (or not apply) to each of the category. The terminology and SC1, SC2 and SC3 categories are discussed in GEH LTR NEDC-33934P/NEDO-33934, Revision 0, BWRX-300 Safety Strategy Licensing Topical Report.
8 – Electric Power	10 CFR Part 50, Appendix A, GDC 1, 2, 3, 4, 5, 17, 18	In Chapter 8, references to certain International Electrotechnical Commission (IEC) standards are made in addition to the IEEE standards. The PSAR should discuss conflicts, if any, between the application of IEC standards and the applicable guidance of IEEE standards or NRC regulatory guides.

Chapter 9 – Auxiliary Systems

Section	Basis for Observation/ Comment	Readiness Assessment Observations
9.1 - Fuel Storage and Handling	10 CFR Part 50, Appendix A, GDC 4 and GDC 61	Section 9.1 refers to neutron absorbing materials and associated testing without naming specific materials or test methods. The use of a material that has previously been reviewed and approved by the staff will result in a less complicated review process that does not require the staff to review initial material qualification testing.

		In addition, guidance on an acceptable neutron absorbing material surveillance program is available in NEI 16-03, “Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools.”
9.1 - Fuel Storage and Handling	10 CFR 50.68, GDC 62	1. Missing design specifications of the spent fuel rack and spent fuel pool design. 2. Missing criticality safety analysis. 3. Missing information on the computer codes used for criticality safety analysis. 4. Missing information on what cross section library is used in criticality safety. 5. Missing code benchmarking analysis. The draft section states that the criticality safety analysis [for spent fuel pool] will be provided in the FSAR, draft PSAR Chapter 9.1, Revision B, Page 9-33, fifth paragraph). This is inconsistent with 10 CFR 50.68 which requires construction permit holders comply with 10 CFR 50.68 (b) or 10 CFR 70.24.
9.3 - Process Auxiliaries	10 CFR 50.62(c)(4)	Section 9.3.5 identifies NEDC-33912-P-A, "BWRX-300 Reactivity Control" as approving the removal of the Standby Liquid Control System, however this approved LTR did not approve an exemption. The SER noted, "The staff concludes that an analysis that demonstrates P(ATWS) is less than 1x10-5 per reactor year [without the use of an SLCS] could support an exemption from 10 CFR 50.62(c)(4). When the NRC receives an application for a BWRX-300, the staff will conduct an evaluation of reliability or probabilistic analysis that demonstrates the P(ATWS) criterion is met [without the use of an SLCS], conforms with Limitation and Condition 5.1 of this SE, and confirms that special circumstances justify an exemption from 10 CFR 50.62(c)(4)." No exemption or mention of an exemption are listed in Section 9.3.5, nor an evaluation pursuant to Limitation and Condition 5.1 of NEDC-33912-P-A, "BWRX-300 Reactivity Control."
9.5 - Other Auxiliary Systems	SRP 9.5.8	Section 9.5.8 does not discuss the location of air intake in relation to SDG exhaust in order to prevent dilution or contamination.
9.5 - Other Auxiliary Systems	GDC 2	Section 9.5.4 references GDC 2 but does not provide detail on how the system meets the GDC, such as designation as seismic Category I. P&IDs have historically been used as an aid to show where

		seismic/quality classification changes occur within a system, if the entire system is not a single classification
9.5 - Other Auxiliary Systems	Clarity	The text in Section 9.5.1.2.6, subsection “Fire Pumps,” identifies where two of the three fire pumps are located. It is unclear where the second electric fire pump is located.
9.5 - Other Auxiliary Systems	Clarity	In Section 9.5.1.2.6, it is unclear whether the “Fire Water Enclosure” identified for an electric fire pump or the unnamed “fire-rated enclosure” for the diesel-driven fire pump are the same structure as the “Fire Pump House” identified on Figure 9.5-1.
9.5 - Other Auxiliary Systems	Clarity	Section 9.5.1.6 identifies RG 1.189, Revision 4 as applicable guidance. The staff notes that Revision 5 to RG 1.189 has been issued.
9.5 - Other Auxiliary Systems	Clarity/RG 1.189	NFPA standards 16, 92A, and 1081 are mentioned in in the text of Section 9.5.1, but do not appear in Table 9.5-1, “List of Applicable Codes, Standards and Regulatory Guidance for Fire Protection.”
9A - Fire Hazards Analysis	RG 1.189 Section 8.2	The strategy described in Section 9A.3.2 seems to imply that SSCs within the fire area where the fire occurs may be relied on to achieve safe shutdown. This would be contrary to the guidance in RG 1.189, Section 8.2, but no deviation from the guidance has been identified.

Chapter 10 – Steam and Power Conversion System

Section	**Basis for Observation/Comment**	**Readiness Assessment Observations**
10.4 – Other Features of Steam and Power Conversion System	10 CFR Part 50, Appendix A, General Design Criteria 4	Section 3.1.1.4, “GDC 4 – Environmental Dynamic Effects Design Bases,” states that for piping in containment penetration areas, the probability of fluid system rupture is demonstrated to be extremely low under conditions consistent with the design basis piping, such that line breaks do not need to be postulated for the purpose of evaluation of potential dynamic effects. (a) 10 CFR 50, Appendix A, GDC 4 is for structures, systems, and components important to safety and is not limited to “piping in containment penetration areas.”

		(b) The text does not reference a document demonstrating that the probability of fluid system rupture is extremely low under conditions consistent with the design basis piping.

Chapter 11 - Radioactive Waste Management

Section	**Basis for Observation/ Comment**	**Readiness Assessment Observations**
11 – Radioactive Waste Management	10 CFR 20.1301 10 CFR 50.34a	The draft PSAR references the ESP for doses to the public in Section 11.3.3 and in other sections. However, the BWRX-300 was not included in the ESP and some of the radionuclides releases in draft PSAR Tables 11.2-3 and 11.3-2 are higher than the radionuclide releases in the ESP Tables 11.2-4 and 11.3-3 respectively. Any deviations from ESP doses need to be justified.
11 – Radioactive Waste Management	RG 1.143	For the components classified in accordance with RG 1.143, the classifications aren't entirely clear. Draft PSAR Section 11.2.2.2 states that the RWB is seismic category RW-IIa and that the components within the RG 1.143 boundary are seismic RW-IIa. Draft PSAR Section 11.4.2.2 states that the spent resin tank, sludge tank, pumps, valves, and piping are classified as RW-IIa. However, while Section 11.3 mentions RG 1.143, it doesn't appear to provide the RG 1.143 classification for the gaseous waste management system. In addition, when section 11.2.2.2 states that the RWB and liquid radwaste components are seismic RW-IIa, it is unclear if that means that they are only meeting the seismic criteria for the RG 1.143, RW-IIa classification or all of the RG 1.143, RW-IIa criteria. In addition, when alternatives to RG 1.143 are used (for example, alternatives are mentioned in Section 11.2.2.2) the alternative code or standard should be specified in the PSAR.

Chapter 12 – Radiation Protection

Section	Basis for Observation/ Comment	Readiness Assessment Observations
12.2 – Radiation Sources	GDC 61	Some source terms appear to be missing from Section 12.2, such as the sources in the LWM filter skid. In addition, some of the equipment parameters and assumptions for developing source terms do not appear to be included in the application. For example, decontamination efficiencies of the LWM filter skid components and dimensions of the size of the skid or its components are not provided; holdup times, dimensions, and capacities of offgas system components do not appear to be specified.
12.3 – Radiation Protection Design Features	10 CFR 20.1502	While draft PSAR Section 11.5 and Table 11.5-1, “PRM Instrument Characteristics” provided information for the specific PRM monitors, no specific area or airborne radiation monitors are discussed in Chapter 12 (other than a discussion of the containment monitoring subsystem in Section 12.3.4.2) and no list of monitors or monitor locations are provided. There is no explanation provided for the staff to understand the reason for the difference and the PSAR should contain an appropriate explanation. ..

Chapter 13 – Conduct of Operations

Section	Basis for Observation/Comment	Readiness Assessment Observations
13.1 -Organizational Structure	50.34(a)(6) - A preliminary plan for the applicant's organization, training of personnel, and conduct of operations. 50.34(a)(9) - The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter. 1) NUREG-0800, Section 13.1.1, “Management and Technical	1) While organizational experience is discussed in the draft PSAR, there is no discussion of the role and use of risk insights in risk-informed evaluations and decision-making within the project. 2) While several corporate officer positions are described in the draft PSAR (e.g., President/CEO, Senior Vice President, New Nuclear Projects, etc.), there is no associated commitment made regarding a specific corporate officer with responsibility for nuclear activities that will have no ancillary responsibilities that might detract attention from nuclear safety matters.

	Support Organization" (Revision 6), Criterion III.1(a)(vii), states "As demonstrated on organizational charts, in descriptions of organizational functions and responsibilities, and in descriptions of position functions and responsibilities, the applicant…has described how the organization will carry out its responsibilities to control major contractors and has committed to consider safety first, with due consideration of risk insights, in design and construction of the facility and during the transition from construction through testing to operation…" 2) NUREG-0800, Section 13.1.1, Criterion III.1(b), states "The corporate officer responsible for nuclear activities should be identified and a commitment made by the applicant that this individual will have no ancillary responsibilities that might detract attention from nuclear safety matters." 3) NUREG-0800, Section 13.1.1, Criterion III.1(d), states "Management and organizational responsibilities are clearly defined to address HFE considerations." 4) NUREG-0800, Section 13.1.1, Criterion III.1(e), states "The organizational units involved in the design and construction of the project communicate among each other in a searchable and retrievable documented form, and management clearly and unambiguously controls the project and its documentation. Clear management control and effective	3) Discussion of the HFE component of the organization, including its management and organizational interface, is not discussed in draft PSAR Chapter 13. Additionally, the staff expect that the PSAR will address the application of the state-of-the-art human factors principles to the control room design in a manner consistent with 50.34(f)(2)(iii), including what standards will be applied during the design process to achieve that. 4) The draft PSAR does not describe how the communications between the organizational units involved in the design and construction of the project will be captured in a searchable and retrievable documented format.

	lines of authority and communication exist among the organizational units involved in managing, operating, and providing technical support for the facility. There is clear management control of the organizational units involved in operating and providing technical support for the facility, and there are clear lines of authority between management and these groups and effective communication among them and with management.”	
13.2 - Training	50.34(a)(6) - A preliminary plan for the applicant's organization, training of personnel, and conduct of operations. 50.34(a)(9) - The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter. 1) NUREG-0800, Section 13.2.1, “Reactor Operator Requalification Program and Training” (Revision 4), Criterion III.1(a) states “The applicant has committed to RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants.” RG 1.8 endorses American National Standards Institute /American Nuclear Society (ANSI/ANS)-3.1-1993, “Selection, Qualification, and Training of Personnel for Nuclear Power Plants.” Criterion III.1(e)(1) states “For nuclear power plant license applicants, the technical submittal shall demonstrate that a licensed operator training program will be established implemented, and	1) The NRC staff note that the current revision of RG 1.8 no longer addresses licensed operator qualifications and training. Specifically, RG 1.8 revision 4 notes that “…the NRC removed the applicability of 10 CFR Part 55 from this revision of RG 1.8 so that NRC guidance for operator license qualifications will be located solely in NUREG-1021, which references the [National Academy for Nuclear Training (NANT)] qualification standards.” The PSAR should describe how this is area will be addressed in light of this change. The NANT qualification standards and, more broadly, training program accreditation by the Institute of Nuclear Power Operations (INPO), represent one method that the NRC recognizes for meeting these qualification and training program criteria; facility license applicants may propose other approaches as well. Facility licensees seeking to use alternatives to National Nuclear Accrediting Board (NNAB) accreditation should submit their initial licensing training program to the NRC operator licensing program office for review and acceptance as a Commission-approved training program. 2) The NRC staff note that the draft PSAR does not Reference NEI 06-13A or any other approach for cold-plant operator licensing.

	maintained by 18 months prior to fuel load by means of the following …The applicant has described how the licensed operator training program conforms to RG 1.8, “Qualification and Training of Personnel for Nuclear Power Plants…” Criterion III.1(e)(2) states “For nuclear power plant license applicants, the technical submittal shall demonstrate that a licensed operator training program will be established implemented, and maintained by 18 months prior to fuel load by means of the following …The subjects covered in the licensed operator training program should include, as a minimum, the subjects in 10 CFR 55.31, ‘How to Apply,’ 10 CFR 55.41, ‘Written examination: Operators’; 10 CFR 55.43, ‘Written examination: Senior operators’; 10 CFR 55.45, ‘Operating tests’; and RG 1.8 for reactor operators and senior reactor operators, as appropriate…” 2) NUREG-0800, Section 13.2.1, “Reactor Operator Requalification Program and Training” (Revision 4), Criterion III.1(b) states that “The… applicant has committed to NEI 06-13A, “Template for an Industry Training Program Description.” NEI 06-13A describes a training program that the staff has found as a way to describe an acceptable licensed operator training program.” Criterion III.1(e)(3) states “For nuclear power plant license applicants, the technical submittal shall demonstrate that a licensed	

	operator training program will be established, implemented, and maintained by 18 months prior to fuel load by means of the following… the applicant has described how the licensed operator training program conforms to NEI 06-13A, “Template for an Industry Training Program Description.” NUREG-0800, Section 13.2.2, “Non-Licensed Plant Staff Training” (Revision 4), Criterion III.1(b) states that “The… applicant has committed to NEI 06-13A, ‘Template for an Industry Training Program Description.’ NEI 06-13A describes a training program that the staff has found as a way to describe an acceptable non-licensed plant staff training program.” Criterion III.1(c)(ii) states that “For nuclear power plant applicants, the technical submittal shall demonstrate that the non-licensed plant staff training program will be established, implemented, and maintained by 18 months prior to fuel load by means of the following… the applicant has described how the non-licensed plant staff programs conform to NEI 06-13A.”	
13.3 – Emergency Preparedness	10 CFR 50.34(a)(10)	TVA should identify and justify ESP action items for dispensation at the OL application stage. The draft PSAR information identified permit conditions and action items but did not justify the reason for deferring them to the OLA.
13.5 - Plant Procedures	50.34(a)(6) - A preliminary plan for the applicant's organization, training of personnel, and conduct of operations. 50.34(a)(9) - The	1) The draft PSAR does not describe how administrative procedures are provided for evaluating operating, design and construction experience to ensure that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.

	technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter. 1) NUREG-0800, Section 13.5.1.1, “Administrative Procedures – General” (Revision 2), Criterion III.1(g) states “Administrative procedures to provide feedback on operation, design, and construction of the facility should comply with 10 CFR 50.34(f)(3)(i) and with NUREG-0737, “Clarification of TMI Action Plan Requirements,” Task Action Plan Item I.C.5.” 50.34(f)(3)(i) states “Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.” 2) NUREG-0800, Section 13.5.1.1, Criterion III.1(h), states “Administrative controls governing crane operations must include a requirement that the operators of cranes over fuel pools be qualified and conduct themselves in accordance with the guidelines of ANSI-B30.2-1976, "Overhead and Gantry Cranes." 3) NUREG-0800, Section 13.5.1.1, Criterion III.1(i), states “A vendor interface program should ensure that vendor information for safety related components is incorporated into plant documentation as described in Generic Letter (GL) 90-03,	2) The PSAR does not discuss whether the administrative controls governing crane operations will include a requirement that the operators of cranes over fuel pools be qualified and conduct themselves in accordance with the guidelines of ANSI-B30.2-1976, "Overhead and Gantry Cranes." 3) The draft PSAR does not discuss whether a vendor interface program will ensure that vendor information for safety related components is incorporated into plant documentation as described in Generic Letter (GL) 90-03, ‘Relaxation of Staff Position in Generic Letter 83-28.’”

	‘Relaxation of Staff Position in Generic Letter 83-28.’”	
13.–4 – Operational Programs	Per 10 CFR 73.55(a)(4), the requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage shall be implemented before fuel is allowed onsite (protected area).	Observation for Table 13.4-101 in the pre-application activities. Instead of “Fuel Receipt” for the Implementation Milestone, it should be “Prior to receipt of fuel onsite (protected area),” in accordance with 10 CFR 73.55(a)(4). Although the SRP states “Fuel Receipt” this has been changed for recent design centers (e.g., AP1000 and ESBWR) as “Prior to receipt of fuel onsite (protected area)” was determined to be less ambiguous.
13.6 - Security	10 CFR 73.56 and 73.57 SRP 13.6.4, Table 1	1.The text added after the 8th paragraph includes, "the construction access program will provide for transition of procedures to operational authorize measures of 10 CFR 73.56 and 73.57", but it is unclear how the transition into the operational program would be implemented. This should be clarified in the PSAR. 2. The licensee should consider guidance in Table 1 SRP 13.6.4 for determining AA measures prior to fuel in the protected area, while transitioning into their operational program.
13.6 - Security	10 CFR 50.54(p) 10 CFR 73.21 10 CFR 73.22 2.390 10 CFR 50.34(c)	The text added after the 8th paragraph on page 38 mentions that the implementing procedures, including testing and maintenance, will be developed and maintained in accordance with 10 CFR 50.54(p). The PSAR should clarify the use of 10 CFR 50.54(p) as it relates to these implementing procedures. Also on page 38, the PSAR should add the regulation for 10 CFR 73.22 in addition to 10 CFR 73.21 for SGI handling. The PSAR should remove the reference of 10 CFR 73.21 in the sentence for the handling of security-related information (SRI). The requirements of 10 CFR 73.21 and 10 CFR 73.22 are designated for SGI (and SGI-M for 10 CFR 73.21 and 10 CFR 73.23). SRI should fall under the requirements of 2.390.
13.7 - Fitness for Duty	10 CFR 26.4, “FFD program applicability to categories of individuals.”	The staff observes that NRC endorsed NEI 06-06, Revision 6, only provides one acceptable method for FFD program guidance for individuals categorized in 10 CFR 26.4(f) and not for individuals categorized in 10 CFR 26.4(e), (g), etc. Consequently, bullets 2 through 5 listed under the Chapter 13.7 paragraph that starts with "The construction phase FFD program is consistent with NEI 06-06 . . . " should be moved to a new (i.e., next) paragraph that essentially clarifies that certain individuals who perform certain roles and responsibilities, not listed in 10

	RG 5.84, “Fitness-For-Duty Programs at New Reactor Construction Sites” RG 5.89, "Fitness-for-Duty Programs for Commercial Power Reactor and Category I Special Nuclear Material Licensees."	CFR 26.4(f), but who are subject to Part 26, will be subject to an FFD program that meets the requirements described in 10 CFR 26.4(e), (g), and (h) as applicable. Also, if PSAR does not commit to or implement all the guidance in RG 5.84 or 5.89, then it should describe those major technical areas that propose to implement alternative guidance to meet the FFD rule.
13.7 - Fitness for Duty	10 CFR 26.4 RG 5.84, “Fitness-For-Duty Programs at New Reactor Construction Sites” RG 5.89, "Fitness-for-Duty Programs for Commercial Power Reactor and Category I Special Nuclear Material Licensees." NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” Staff Comment: Table 13.4-101, "Operational Programs and Implementation Milestones," states that the "FFD Program for Operation" has a milestone at "Fuel Receipt in the Protected Area." This is not correct and should be updated to reflect the requirements in 10 CFR 26.4; same comment applies to persons who report to the TSC and EOF.	The PSAR should refer to the requirements in 10 CFR 26.4 as summarized in NEI 06-06, Table 3-1, "FFD Program Applicability and Milestones—For Construction and Transition to Operations," or NUREG-0800, Chapter 13.7.2, Table 1 - "FFD Program Applicability and Milestones," to ascertain when an FFD program for operation must be implemented. For example, FFD Program for persons required to physically report to the Technical Support Center or Emergency Operations Facility applicability applies to individuals "Prior to the conduct of the first full participation emergency preparedness exercise under 10 CFR Part 50, Appendix E, Section F.2.a," not as stated in the Table 13.4-101, "Initial Operations." Additionally, the milestone for FFD Program for Operation is inconsistent with 10 CFR 26.4. The PSAR should use the correct milestones descriptions in NEI 0606, Table 3-1.
Chapter 14 – Initial Test Program		

Section	Basis for Observation/ Comment	Readiness Assessment Observations
14.1.1 - Scope of Test Program	10 CFR Part 50, special treatment requirements	1. The PSAR should specify the scope of the Initial Test Program (ITP) to include safety-related systems and components, and any systems or components not classified as safety-related but within the scope of special treatment requirements of 10 CFR Part 50. 2. The PSAR should specify that the nuclear power plant will be designed and constructed to accommodate the necessary preoperational and startup testing. The preoperational and startup testing required should be a consideration early in the design process. It is not an expectation that detailed, step-by-step written procedures are readily available at the CP stage. However, the PSAR should provide a high-level description of the types of testing to be conducted to ensure that significant testing is not overlooked in the design and construction process.
14.1.2 - Plant Design Features That Are Special, Unique, or First of a Kind	10 CFR 50.43(e)	1. The PSAR should discuss the plans to ensure that the preoperational and/or startup testing for the Isolation Condenser System (ICS) and ICS Condensate Return to the Chimney will be correlated to the 10 CFR 50.43(e) testing and the system/component qualification testing. 2. The PSAR should discuss preoperational testing plans for Reactor Pressure Vessel Isolation Valves in BWRX-300. 3. The PSAR should provide a detailed description of the BWRX-300 design specifying design features that are first of a kind (FOAK) and need to meet 10 CFR 50.43(e). 4. The PSAR should describe the plans to consider the potential for large thermal gradients during startup testing that can exceed ASME BPV Section III stress limits or cause excessive fatigue usage due to the low flow, low decay heat conditions.
14.1.4 - Utilization of Plant Operating and Testing Experiences at Other Reactor Facilities	RG 1.70	As noted in RG 1.70, the PSAR should describe the schedule for conducting a study of available information on reactor plant operating experiences and implementing the program. The PSAR should describe the plans to address the safety issues described in NRC generic communications, such as power-operated valve capability.

14.1.5 - Test Program Schedule	General – Review schedule	The NRC staff will need to review the test procedures as part of the Operating License (OL) application review and might need more than 60 days (as indicated in the PSAR) to review the procedures and provide feedback.

Chapter 15 – Safety Analysis

Section	**Basis for Observation/ Comment**	**Readiness Assessment Observations**
15 – Safety analysis	Information/Clarity	Section 15.2.2 of the PSAR discusses categorization of events according to their frequency of occurrence. The section mentions that selection of event frequency is described in Subsection 15.6.3.1. However, justification or documentation for selection each of the events analyzed in Section 15.5 to their assigned classification is not provided. For example, Section 15.5.3.1.1 analyzes Loss of Feedwater Heating event as a BL-AOO while 15.5.4.1.1 has Loss of All Feedwater heating as a CN-DBA. Similarly, Section 15.5.3.2.1 analyzes Generator Load Rejection or Turbine Trip as BL-AOO and Section 15.5.4.2.1 has the same event, with passive common cause failure of the DL2 functions, analyzed as CN-DBA. The applicant should provide a detailed evaluation for selection of each of the events analyzed in Section 15.5 to their respective assigned event categories. For example, in some cases these event classifications do not coincide with the ESBWR event classifications.
15 – Safety analysis	Regulatory Guide (RG) 1.70, “Standard Format and Content of Safety Analysis Report for Nuclear Power Plants	Chapter 15 of the PSAR lists various LTRs including BWRX-300 Safety Strategy and other LTRs for various codes (e.g. TRACG, PANAC11, PAVAN etc.) and methodologies. Regulatory Guide (RG) 1.70, “Standard Format and Content of Safety Analysis Report for Nuclear Power Plants,” provides guidance regarding which material must be incorporated by reference. The PSAR Chapter 15 should also disposition all the limitation and conditions associated with the LTRs incorporated in the Chapter 15 safety analyses.
15.5 - Deterministic Safety Analysis	10 CFR 50.2, 10 CFR Part 50, Appendix A	Section 15.5.3 of the PSAR presents analysis of the Anticipated Operation Occurrences (AOOs). The analysis of events listed within the sub-sections of Section 15.5.3 shows DL2 functions being credited to mitigate the AOOs. The BWRX-300 Safety Strategy LTR states that that

		SC3, ‘Non-Safety-Related Important to Safety’ and SCN, ‘Non-Safety-Related’ SSCs perform the DL2 primary and support functions while SC1, ‘important to safety’ or ‘safety related’, are required to perform DL3 primary and integral support functions. Based on the description provided in the BWRX-300 Safety Strategy LTR, DL2 SSCs are the primary success path for protecting multiple fission product barriers (e.g., SAFDLs, RCPB) during design-basis transients (i.e., AOOs). The NRC staff notes that AOOs are considered design-basis events, and consistent with the definition of “safety-related” in 10 CFR 50.2, any SSCs that perform the functions prescribed in the definition are required to be classified as “safety-related." The 10 CFR 50.2 states that safety related SSCs are those that are relied on during or following a design basis event to assure, in part, (1) The integrity of the reactor coolant pressure boundary, and (2) The capability to shut down the reactor and maintain it in a safe shutdown condition. AOOs are defined in Appendix A to 10 CFR Part 50, as those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit. The general design criteria (GDC) in 10 CFR Part 50, Appendix A provides the minimum requirements and criteria for maintaining the integrity of the reactor coolant pressure boundary (RCPB), and for shutting down the reactor and maintaining it in a safe condition for AOOs and postulated accidents such that there is reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. For AOOs, the GDCs prescribe a safe shutdown condition to be one where decay heat is being sufficiently removed and the fuel integrity barrier is maintained by demonstration of appropriate margin to the specified acceptable fuel design limits (SAFDLs) (e.g., GDC 34). The NRC staff notes that classification of the entire DL2 as non-safety related places significantly more reliance on non-safety related equipment to mitigate AOOs and will require appropriate justification. If it’s determined an exemption is needed for use of non-safety related SSCs to mitigate AOOs, that should also be included in the PSAR.
15.5 - Deterministic Safety Analyses	10 CFR 50.12, 10 CFR Part 50, Appendix A	The deterministic safety analysis performed in Section 15.5 of the PSAR shows that the Single Failure criterion is only applied to DL3 Safety Class 1 (SC1) SSCs. This is consistent with the BWRX-300 Safety Strategy LTR which states that meeting single failure criterion is not a design rule applied to Safety Category 2 or 3 functions. The LTR states that redundancy may be required for other reasons and points out that

		sufficient mechanical, electrical and instrumentation component should be provided. The staff notes that 10 CFR Part 50, Appendix A, General Design Criteria require that the design include the capability to withstand single failures. These requirements are not exclusive to design-basis accidents. Several GDCs have requirements for SSC reliability for mitigation of AOOs (e.g., GDC 17, 21, 22, 25, and 34). Alternate approaches to application of Single Failure criterion need to provide appropriate justification in the PSAR and may require exemptions to applicable regulatory requirements in accordance with 10 CFR 50.12.
15.5 - Deterministic Safety Analyses	Information/Clarity	The input parameters used in the analysis for the AOOs, and the DBAs are presented in Table 15.5-3. Based on the information provided in the Table, it appears to the NRC staff that nominal values were used as an input for the various plant parameters for the analyses performed and no information is provided on the biasing of the values. The PSAR should include justification on the use of nominal values for the subject analyses or provide details on the conservative biasing used for each of the input parameters. For example, Section 15.5.3.2.3 of PSAR provides AOO analysis of loss of condenser vacuum. However, the Table 15.5-3 does not contain the bounding conservative value for initial rated core power considering measurement uncertainties, feedwater flow rate and feedwater runback coast down time and initial MCPR of the hot channel.
15.5 - Deterministic Safety Analyses	10 CFR 50.46, 10 CFR Part 50, Appendix A, 10 CFR 50.12	The deterministic safety analysis discussed for the LOCA events in Chapter 15.5 of the PSAR does not provide any evaluation for losses of coolant at the reactor isolation valve to the reactor vessel connections. [LOCAs are defined in 10 CFR 50.46(c) and 10 CFR Part 50, Appendix A, and analyzed in SAR Chapter 15 as non-mechanistic hypothetical breaks to establish the design-basis of the ECCS. 10 CFR 50.46(a)(1)(i) states, “ECCS cooling performance must be calculated…for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated.” The PSAR should address losses of coolant from between the RPV assembly and RPV isolation valve assemblies. If these LOCAs are excluded from the design-basis of the ECCS, the PSAR should justify, in terms of both likelihood and consequences, the basis for such a determination including supporting realistic thermal hydraulic analysis demonstrating sufficient

		core cooling. If it's determined an exemption is needed, that should also be included in the PSAR]
15.5 - Deterministic Safety Analyses	NUREG-0800, Chapter 15, 10 CFR 50, Appendix A	Section 15.5.3.2.2 of the PSAR lists closure of One Main Steam Reactor Isolation Valve (MSRIV) as an BL-AOO event. Section 15.5.5.2.1 of the PSAR later lists the closure of one MSRIV event as an EX-DEC event. The postulated initiating event for the EX-DEC one MSRIV closure is the same as the BL-AOO event but assumes a common cause failure hydraulic scram failure. The NRC staff notes that one main steam line isolation valve closure event is listed in the SRP (NUREG-0800) Chapter 15 as events that occur with moderate frequency and have been analyzed for other BWRs as DBEs, including the ESBWR DCD which lists it as an AOO. Please provide justification on classification of event of moderate frequency like closure of one MSRIV as an EX-DEC event.
15.5 - Deterministic Safety Analyses	10 CFR Part 50, Appendix A, 10 CFR Part 100	The safety analysis performed in Chapter 15.5 of the PSAR does not include the inadvertent loading and operation of fuel assembly in an improper position. Analysis of this event is required to meet the acceptance criteria for GDC 13, pertaining to providing instrumentation to monitor variables over anticipated ranges for normal operations, for anticipated operational occurrences, and for accident conditions, as well as compliance to 10 CFR Part 100, as it relates to offsite consequences resulting from reactor operations with an undetected misloaded fuel assembly. The event is listed in SRP (NUREG-0800), Chapter 15.4.7. As an example, the NRC staff notes that ESBWR DCD lists Fuel Assembly Loading Error, Mislocated Bundle as Infrequent Event. Infrequent Event is defined as the DBE with probability of occurrence < 1/100 and radiological consequence less than a design basis accident (≤ 2.5 rem). The ESBWR DCD states that the GESTAR analysis (GESTAR II, Amendment 28) for Fuel Assembly Loading Error, Mislocated Bundle bounds the ESBWR design. For the event, the DCD also lists that "Should a bundle mislocation, misorientation, and improper seating occur and go undetected, the plant specific acceptance of the generic GESTAR analysis is revoked, and the classification of this event is changed from "infrequent incident" (infrequent event) classification to an "incident of moderate frequency" (AOO) classification immediately for that plant." The PSAR needs to provide justification for not analyzing the inadvertent loading and operation of fuel assembly in an improper position event in the deterministic safety analyses.

15.5 - Deterministic Safety Analyses	Information/Clarity	Section 15.5.5.1 of the PSAR states that the NRC approved Control Rod Drive Accident (CRDA) methodology (LTR NEDE-33885-P-A, Revision 1) will be applied to the BWRX-300 to demonstrate that the cladding failures do not occur for postulated CRDA. It further states that results of rod drop calculations will be discussed in the PRA which will be summarized in a future licensing activity. The PSAR cites L&C 5.2 of the SER to the CRDA methodology and states that 'An exemption is provided in Section 3.1 to justify the CRDA as practically eliminated event with accompanying probabilistic risk assessment'. The NRC staff notes that the portion of Section 3.1 of the PSAR describing this exemption is not available at this time and NRC staff cannot provide any feedback on Technical Basis and Probability Analysis provided in subsections of Section 15.5.5.1 in absence of the exemption request along with the PRA.
15.5 - Deterministic Safety Analyses	Information/Clarity	Section 15.5.2 of the PSAR provides the stability analysis results regarding the core wide mode of density wave oscillations. It is concluded that the system decay ratio is less than the acceptable limit for normal power operation. Although BWRX-300 operating domain is different from operating BWRs due to the in-vessel natural circulation, the stability analysis should reflect the possible range of RPV pressure, feedwater temperature and potentially the downcomer water level permitted for operation.
15.5 - Deterministic Safety Analyses	Information/Clarity	Section 15.5.3.2.2 provides the analysis results of an AOO of the Closure of One Main Steam Reactor Isolation Valve. Section 15.5.4.2 provides the analysis results of a DBA of the closure of all MSRIVs and FWRIVs. The listing of DBAs appears to be missing an event for the simultaneous closure of all MSIVs (without FWRIV closure), which could be more severe if the closure of the FWRIVs is not assumed. Provide justifications that the closure of all MSIVs is not a DBA event or the relevant analysis results of all MSIVs closure as a DBA.
15.5 - Deterministic Safety Analyses	Information/Clarity	Section 15.5.4.5 states that acceptance criteria for fuel integrity for LOCA DBA is demonstrated by showing that the water level does not fall below the Top of Active Fuel (TAF), or the fuel cladding temperature does not exceed the cladding temperatures during normal operation. However, the results presented for LOCA in Table 15.7-3 only list peak cladding temperatures and do not provide information on water level. The water level results for each of cases listed should be included.

15.5 - Deterministic Safety Analyses	50.34(a)(1)(ii)(D) and RG 1.183	There appears to be an inconsistency in the cited references to the RADTRAD versions used in the calculation of radiological consequences. Section 15.5.1.2.6 RADTRAD, states that “The dose consequences of postulated design basis accidents are calculated using the RADTRAD Version 3.10 computer code (Reference 15.5-11). Reference 15.5-11 is listed as NUREG/CR-7220, SNAP/RADTRAD 3.10: Description of Models and Methods, June 2016. However, the title of NUREG/CR-7220 is SNAP/RADTRAD 4.0: Description of Models and Methods. Note: RADTRAD 3.10 was translated into JAVA from Fortran to provide the base program that became SNAP/RADTRAD. Section 15.5.9.1 Analysis of LOCA Outside Containment states that dose consequence analyses are performed using NUREG/CR-6604 RADTRAD Version 3.10. NUREG/CR-6604 describes the original RADTRAD code version 2.20 1997. Clarification as to the version of RADTRAD used to compute design basis accident dose consequences should be provided.
Table 15.6-1 - Probabilistic Safety Assessment Objectives Table 15.7-9 - Core Damage Frequency Results Table 15.7-10 - Large Release Frequency Results Section 15.6.9 - Results of the Level 1 Probabilistic Safety Assessment, Section 15.6.9.1.4 - Large Release Frequencies,	10 CFR 50.34, 50.2, 50.36, Appendix A, Severe Accident Policy Statement, Safety Goal Policy Statement, SRP Chapters 15 and 19.3	**Overall Observations** - The NRC staff identified major gaps in the content of information in the areas of key risk insights and quantitative results for the design, dominant accident sequences, design features that are major risk contributors, and quantified risk metrics. If not addressed, based on the uses of the PRA and to the extent that the PRA supports other portions of the application (e.g., classification of licensing basis events), these gaps have the potential to challenge an effective and efficient review of the CP application, including, but not limited to, reaching positive regulatory findings on (1) implementation of the Safety Strategy, particularly the uses of the PRA in the methodology; (2) achievement of the stated objectives of the PRA in Table 15.6-1 of the PSAR; (3) meeting the Commission’s Safety Goals and, consequently, the assurance of no undue risk to public health and safety (objective 4 of the PRA in Table 15.6-1); (4) meeting the Commission’s containment performance goals (probabilistic and deterministic) as described in SECY 90-016 and SECY 93-087 and associated SRMs; (5) conformance with the Commission’s Severe Accident Policy Statement; (6) conformance with the Commission’s direction on the Regulatory Treatment of Non-Safety Systems as described in SECY 94-084 and SECY 95-132 and associated SRMs; and (7) support findings against applicable regulations such as 10 CFR 50.34 (a)(1)(ii) (“…extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products”).

Section 15.6.10.1 - Summary		Specific examples of these gaps are identified below. If the preliminary nature of this information is the reason for the exclusion, appropriate qualifiers can be used in the PSAR by the applicant and in the SER by the NRC staff (e.g., “preliminary”).
Table 15.6-1 - Probabilistic Safety Assessment Objectives	As above	A clarification on why Table 15.6-1 does not include using the PSA as described in the LTR NEDC-33934P, “BWRX-300 Safety Strategy” needs to be provided.
Table 15.6-1 - Probabilistic Safety Assessment Objectives	As above	Objective 6 of Table 15.6-1 states that the PSA is used to identify facility vulnerabilities and systems for which design improvements or modifications to operational procedures could reduce the probabilities of severe accidents or mitigate their consequences. With the understanding that procedures may not be developed at this stage, the draft PSAR lacks examples of design changes made based on risk information and insights to achieve the stated objective. Appendix 15B briefly discusses risk reduction included as Defense Line 4 functions for mitigating design extension conditions. The CP application needs to include clarification on whether this section is intended to represent risk reduction measures based on risk insights from the PSA and provide specific examples of the risk reduction achieved or whether these measures were based on deterministic analysis/defense in depth.
15.6.3.5 - Event Sequence Frequency Quantification 15.6.10.1 - Summary	As above	The draft PSAR lacks a list, with summary description, of dominant sequences for CDF and LRF for the full-power internal events PRA model.
15.6.7 - Uncertainty and Sensitivity Analysis 15.6.7.2.7 - Task Outputs and Preliminary Results	As above	The draft PSAR section is lacking the insights from the full-power internal events PRA such as a list of key assumptions and sources of uncertainty, including design features and design assumptions, impacting the application and a list of sensitivity analyses performed to address the assumptions.
Table 15.7-9 - Core Damage Frequency Results	As above	There are no CDF and LRF results. Furthermore, Table 15.7-9, “CDF Results” and Table 15.7-10, “LERF Results,” are intentionally left blank and state that the tables will be populated in the FSAR. The staff notes that Section 3.2.7.3.4 of the GEH Safety Strategy LTR states that “…a demonstration that BWRX-300 risk metrics…is provided.”

Table 15.7-10 - Large Release Frequency Results		
15.6.3.5 - Event Sequence Frequency Quantification 15.6.10.1 - Summary	As above	The draft PSAR is lacking information of accident sequence analyses including (a) summaries of event trees for each initiating event identified in the initiating event analysis, including a discussion of the sequences for each event tree; (b) a description of necessary and sufficient equipment (safety and non-safety-related) reasonably expected to be used to mitigate initiators; and (c) a description of individual function mission times for each safety function and time windows for each operator action included in the PRA.
15.6.3.4.2 - Fault Trees, Data Analysis	As above	The draft PSAR is lacking information on data analysis including design-specific justification for the failure rates used for first-of-a-kind components.
15.6.3.4 - System Analysis (no dedicated section re: passive safety systems)	As above	The draft PSAR is lacking information on passive safety system reliability for the full-power internal events PRA including: (a) identification of all key thermal hydraulics parameters that could affect the reliability of a passive system and introduce uncertainty into the determination of success criteria and (b) accounted for the uncertainty in the analyses that establish the success criteria.
15.6.3.4.3 - Human Reliability Analysis	As above	Section 15.6.3.4.3 states, "Due to the incompleteness of the design and the lack of procedures, this [PRA] analysis is not expected to meet all the requirements of the standard at this time." With the understanding that this is preliminary information and for a CP application all the requirements of the PRA standard are not expected to be met, the draft PSAR is lacking (a) the identification and description of HFEs that result in initiating events, (b) identification and description of pre- and post-accident HFEs that impact the mitigation of initiating events, (c) identification of any dependent HFEs, and (d) any recovery action credit taken, with justification.
15.6.3.6 - Internal Fire Hazard	As above	For a CP application, either an internal fire PRA or a non-PRA evaluation is recommended. The staff notes that the draft PSAR describes performing an internal fire PSA. Given that an internal fire PSA was performed, the draft PSAR lacks (a) a summary of changes made to the internal events PSA to develop the internal fire PSA addressing each of

		the topics identified previously for internal events PSA and (b) a description of the risk insights.
15.6.3.7 - Internal Flooding Hazard	As above	For a CP application, either an internal flood PRA or a non-PRA evaluation of the risk is recommended. Given that an internal flood PSA was performed, the draft PSAR lacks (a) a summary of changes made to the internal events PSA to develop the internal flood PSA addressing each of the topics identified previously for internal events PSA and (b) a description of the risk insights.
15.6.3.9 - Low Power and Shutdown Probabilistic Safety Assessment	As above	For a CP application, it's recommended low power and shutdown events be evaluated using a PRA or non-PRA method. As described in PSAR Section 15.6.3.9, the applicant developed a Low-Power and Shutdown (LPSD) probabilistic safety assessment (PSA) using ANS/ASME-58.22-2014, the trial use Low Power and Shutdown PRA standard. The LPSD PSA also includes heavy load drops that can cause fuel damage, core damage, and large release. To develop a shutdown model, plant operating states (POSs) were defined in relation to decay heat and the availability of systems. Given that a low power and shutdown PSA was performed, the draft PSAR lacks (a) a summary of the POS analysis, (b) a summary of the systematic identification of potential LPSD initiating events, (c) shutdown event trees, (d) identification of key assumptions used in the evaluation, and (e) the quantitative results and dominant quantified accident sequences.
15.6.4 – Level 2 Probabilistic Safety Assessment	As above	The Draft PSAR lacks the following information regarding Level 2 analysis including (a) event trees and key phenomena for Level 2 PRA and (b) demonstration that the design at the CP-stage meets the Commission's expectations for containment performance for new reactors.
15.6.1.1 - Probabilistic Safety Assessment Scope 15.6.2 - Probabilistic Safey Assessment Overview	As above	Section 15.6.2.3, Internal and External Events and Level 1 Probabilistic Safety Assessment, states "Internal and external events and Level 1 PSA also include: Self-assessment results using NEI 17-01 guidance..." The draft PSAR does not include (a) a description of the self-assessment or peer review, and a summary of any limitations identified by the self-assessment arising from the level of maturity of design and operational details and (b) a description of the applicant's plan for meeting each identified PRA element in the OL PRA. This information provides

15.6.2.3 - Internal and External Events and Level 1 Probabilistic Safety Assessment		confidence to the staff that the applicant has the ability to develop and maintain the PRA post-CP approval to reflect the as-built plant.
15.6.1.1 - Probabilistic Safety Assessment Scope 15.6.2 - Probabilistic Safety Assessment Overview	As above	The draft PSAR lacks a description of a PRA configuration control plan including (a) a description of the process to track assumptions and monitor inputs for PRA and non-PRA evaluations supporting the CP application; (b) a description of how new information will be collected and included in the PRA to maintain the PRA consistent with the as-built, as-to-be-operated plant design; and (c) a description of how reviews of the PRA will be conducted (i.e., self-assessment, peer review, etc.), including the frequency of such reviews.
15.6.3.8.1 - High Wind Hazard 15.6.3.8.3 - Seismic Hazard 15.6.3.8 - Probabilistic Safety Assessment External Hazards	As above	For a CP application, it's recommended external events (e.g., seismic, high wind, and external flood) be evaluated using a PRA or non-PRA method. The staff notes that the draft PSAR describes performing a seismic PRA and high wind PRA. Given that such PRAs were performed, the draft PSAR lacks (a) a summary of changes made to the internal events PRA to develop the seismic PRA and high wind PRA addressing each of the topics identified previously for internal events PSA and (b) a description of the risk insights.
15.6.3.7 - Internal Flooding Hazard Table 15.6-3 - External Hazards Screening	As above	The draft PSAR describes that external flooding was excluded from evaluation. Section 15.6.3.7 states that, "external flooding events are reasonably precluded from the BWRX-300 probabilistic flood analysis based on adherence to the design conditions set forth in the envelope of BWRX-300 standard plant site parameters." Table 15.6-3, "External Hazards Screening," includes an entry "Extreme Rain" that addresses external flooding; however, the screening description for this entry is incomplete. The site-specific basis for screening of the external flooding should be provided in the draft PSAR.
15.6.3.8.1 - High Wind	As above	Section 15.6.3.8.1 discusses high wind risk evaluation and states that, "BWRX-300-specific high wind event frequency and plant effects are applied ... to obtain risk results." In the same section, the applicant also states that, "site-specific data are inputs for the external hazards PSA analyses." It is not clear to the staff whether the high wind PRA

		performed at the CP stage will be based on the BWRX-300 generic high wind data or the site-specific data at Clinch River.
15.6.3.8.3 - Seismic Hazard	As above	In parallel to the above question on the high wind PRA, it is not clear to the staff whether the seismic PRA at the CP stage will be based on the BWRX-300 generic seismic data or the site-specific data at Clinch River. Using site-specific hazard information will provide more realistic risk insights.
15.7 - Safety Analyses Appendix 15A.4 - Reference Source Term for Conditions that are Practically Eliminated Events Table 15A-1 - Practically Eliminated Conditions	As above	Table 15A-1, "Practically Eliminated Conditions" describes many conditions that are practically eliminated. For each practically eliminated condition, the table includes a description of risk reduction design features and design and supporting operating provisions that make each condition "extremely unlikely." This section does not include any quantitative risk-information to support the assertion that the conditions are extremely unlikely. In addition, since during the design stage some of the risk-reduction features may change, the section does not describe any quality control measure to ensure changes to the design/operation would not alter the likelihood of these conditions such that their elimination status would change.
15.7 - Results of Deterministic Safety Analyses and Probabilistic Safety Assessment	50.34(a)(1)(ii)(D) and RG 1.183.	There appears to be a discrepancy between the atmospheric dispersion coefficients for the off-site locations in Section 15.5.8 on page 15-109 and the main control room envelope (MCRE) for a release from the ICS pools and for a Diffuse Source Release from the RB in Section 15.5.9 on pages 15-117 and 15-119 respectively. For these release points the 0-2 hour MCRE atmospheric dispersion factor is lower than the 0-2 hour factor for the EAB. The MCR has no credited filtration. The MCR unfiltered in-leakage is artificially assumed to be 38,000 cfm so the control room airborne concentrations would be essentially the same as the outdoor concentrations. However, Table 15.7-4 indicates that the MCR FHA dose is lower than the EAB dose and roughly proportional to the difference in the X/Q values. Tables 15.7-5, 15.7-6, 15.7-7, and 15.7-8 indicate similar discrepancies These discrepancies do not appear to be credible and should be reconciled.
15.7 - Results of Deterministic Safety Analyses and	50.34(a)(1)(ii)(D) and RG 1.183.	The draft PSAR does not appear to contain the deterministic evaluation of the design basis substantial core melt accident needed to satisfy the requirements of 50.34(a)(1)(ii)(D). The PSAR will need to include an evaluation of this source term or a justification for not including it. **(This**

Probabilistic Safety Assessment		**observation is noted in body of the letter as needing continued engagement between TVA and NRC prior to submittal of the CP application).**
Chapter 17 – Quality Assurance		
17.0 - Introduction	Reference to NEDO-11-209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description."	Section 17.0 states that the QA Program described in Section 17.1 is applicable to the Boiling Water Reactor BWRX-300 standard design activities by GEH. The QA Program for site design, construction, and operation activities controlled by TVA is described in Section 17.5. Given that the TVA New Nuclear QAPD TR does not distinguish between site design and standard design activities, the PSAR should explain how the TVA New Nuclear QAPD TR interfaces with the GEH topical report NEDO-11029-A for the design of the BWRX-300, including which document would apply during the design phase for Clinch River.
17.1 - Quality Assurance During Design and Construction Phases	Reference to NEDO-11-209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description."	1. - NEDO-11029-A, Part I, Section 1.0 states that Part II of this QAPD describes the portion of the Quality Management System (QMS) required to meet the requirements of the USA contained in 10 CFR Part 50, Appendix B, 10 CFR Part 71, 10 CFR Part 21, and ASME NQA-1-2015 Edition. Specifically, it applies to work involving SSCs for nuclear power plants and fuel reprocessing plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. GEH may extend this program to other areas that do not meet these criteria. Since it appears that the GEH has a much narrower applicability of their QAPD to SSCs (i.e., not all safety-related SSCs would be covered by the QAPD), the PSAR should clarify whether TVA intends to expand the applicability of the NEDO-11029-A QAPD to cover all safety-related SSCs and include what criteria will be used to cover non-safety related SSCs of safety significance during the design of the BWRX-300. 2.- Section 17.1 states, "The QA program used for BWRX-300 GEH design activities is based on the standard QA Program documented in topical report NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description," and the additional information in this chapter, which describes and clarifies GEH interfaces and responsibilities with BWRX-300 project participants." Section 17.1.18 states that NEDO-11209-A Section 18 establishes requirements for a system of QA Audits used during design of the BWRX-300. NEDO-11209-A, Section 18.3.1.2 applies different audit frequency requirements depending on whether a

		facility is placed into operation and for activities not related to a licensed facility. The PSAR should describe which of the internal audit frequency requirements from Section 18.3.1.2 would apply to the design of BWRX-300.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	Responsibilities within the D-RAP organization are not clear. In Section 17.4.2, GEH BWRX-300 is described as being responsible for the D-RAP, but no specific job titles and responsibilities are identified. The information provided in the PSAR should be of sufficient detail that it can be verified.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	In Section 17.4.1, the program is described as preliminary, and it is stated that the program will be updated as the design continues. The PSAR should clarify if this is intended to refer to the D-RAP process itself the list of SSCs that result from implementation of the process. This paragraph also discusses the focus on the performance of functions versus systems. The PSAR should clarify if this part of the process is subject to change. The PSAR should clearly identify information that is preliminary, information that is not provided for the construction permit application, and information that sufficiently complete to support the findings and conclusions to be made by the NRC staff.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	In Section 17.4.3.1 on risk significance criteria, the methodology to determine the criteria is not sufficiently described and it is stated that the FV and RAW values may be revised as the PSA evolves. The risk significance criteria should be identified in the PSAR to the extent that design, categorization, and quality assurance requirements rely on the risk significance determination for SSCs. One option is the use of a topical report to propose a methodology for determining risk significance in advance or in parallel with the construction permit application.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	In Section 17.4.3.2, the pre-application material includes the statement, "To address gaps in the PSA, deterministic analysis can be used...." There are other reasons to use deterministic analysis to support the determination of risk significance, such as not to over-rely on risk-based information, and to address inherent limitations of the PSA, especially for an unbuilt design. The deterministic inputs should also be described in additional detail, and the D-RAP process should specify the roles and responsibilities for considering different deterministic inputs.

17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	The composition of the expert panel should be described such that the staff can verify the expertise and experience of the expert panel members.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	In Section 17.4.5, the PSA should not be the only input to the list of risk-significant SSCs. Globally, the description of the D-RAP process should reflect its planned implementation, and the PSA should not be over relied on for risk significance determination. It should be clear, for all individual elements that make up the D-RAP, what inputs are considered and how they are used (e.g., risk-based vs. risk-informed).
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	Section 17.4.2 describes the GEH Organization responsible for D-RAP. Section 17.4.6 describes the RAP will be implemented by TVA during operations. The application should be clear on the responsibilities between these two organizations, especially related to the transition between construction and operational RAP activities.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084, SRP 17.4	The last paragraph before Section 17.4.1 states, "the programmatic elements for developing and implementing the D-RAP (organization, design control, procedures, SSCs identification, and implementation) that will be applied prior the initial fuel load are described." Some of these elements (e.g., design control, procedures) are briefly mentioned under organization, but it is not apparent how these are controlled to support the D-RAP process. There is no description of the corrective action program, records, or audit plans as discussed in SRP 17.4.
17.4 - Reliability Assurance Program During Design Phase	Overarching	The context for the D-RAP/RAP is minimal related to any interfaces with the safety strategy, safety and risk categorization provided in Section 3.2, and to a lesser extent PRA and risk insights determined from Section 15.6. The submittal and review will benefit from an understanding of the complete picture.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	The purposes of the reliability assurance program do not include “minimize the frequency of transients that challenge risk significant SSCs.” Does the D-RAP not intend to follow the purposes as outlined in the SECY papers? The applicant should address any alternative to D-RAP that will address purposes that are not covered by D-RAP

17.4 - Reliability Assurance Program During Design Phase	Overarching	For a construction permit application, it is understood that the final list of D-RAP SSCs may not be necessary for the preliminary safety analysis report (PSAR). The PSAR should describe the plans for D-RAP/RAP such that the staff understands the timeline of the review for CP and OL, including any planned topical reports on this topic.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	It is not clear how the D-RAP will be utilized during the design and equipment specification phases if there is no list of D-RAP SSCs. Further, there is no discussion of whether the D-RAP will be utilized during the construction phase. The PSAR should consider including a discussion of the maintenance/QC of the D-RAP list prior to completion of the detailed PSA.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	Section 17.4.3 states, "A list of SSCs that are associated with the initiation, prevention, detection, or mitigation of any failure sequence and have a significant impact in reducing the possibility of damage to fuel and/or associated release of radionuclides is developed and controlled as a BWRX-300 D-RAP report, following development of the detailed PSA." The PSAR should clarify whether these are these the functions referenced in Section 17.4.1 or how this list of functions relates to the functions that will be identified by the D-RAP expert panel.
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	Does TVA plan to identify regulatory treatment of non-safety systems (RTNSS) SSCS? If so, are these SSCs determined to be risk significant?
17.4 - Reliability Assurance Program During Design Phase	SECY-93-087 and SECY-94-084	The PSAR should include a description of how expert panel decisions are documented.
17.5 – Quality Assurance	PSAIs in the Final SE for the TVA New Nuclear QAPD.	1 - Responses provided to Plant Specific Action Items (PSAIs) from the TVA QAPD (NNP-TR-001) have pointers to other PSAR Sections. PSAIs 4.6 and 4.7 point to Section 1.9. These PSAI references were not available for review. 2 - PSAI 4.2 required the CP to include an organizational diagram that aligns with the description of the roles and responsibilities of the Senior Vice President Engineering Operations Support in Part II, Section 1.2.1 and 1.3.2.2.3 of the TVA QAPD. In Section 17.5, the response to Plant Specific Action Item 4.2 states that "PSAI 4.2 is addressed in Section 17.5". However, there is no additional information provided in 17.5.

		3 - Not all PSAIs identified in the final SE for the QAPD Topical Report were addressed by Chapter 17 of the PSAR. While all PSAIs from the draft SE for the TVA QAPD were identified in the PSAR, the two additional PSAIs added in the Final SER need to be included.
General Design Criteria (GDCs) and Principal Design Criteria (PDCs)		
Section 3.1	General	Section 3.1 of PSAR did not include exemptions to GDCs to replace them with PDCs. If it's determined exemptions are needed, they should also be included in the PSAR.
Section 3.1, PDC 17	GDC 17	The electrical system that provides the safety function to protect the SAFDLs and RCS design conditions during AOOs (anticipated operational occurrences) is not safety related. See first observation under "Chapter 15.5 – Deterministic Safety Analysis"
Section 3.1, PDC 17	GDC 17	PDC 17 - First statement states: An electric power system shall be… PDC 17 - Second statement states: The safety function for each power system... PDC 17 - Second Para states: The electric power system shall meet reliability targets commensurate with the importance of the safety function for each power system. The applicant needs to define how many power systems there will be (such as onsite and offsite) and each power system should be named/listed in PDC 17. PSAR Section 3.1 should identify where the reliability targets of each power system are provided in the PSAR.
Section 3.1, PDC 19	GDC 19 and 50.34(a)(1)(ii)(D)	The staff's expectation is that adequate radiation protection for control room personnel should be demonstrated for all design basis accidents including the deterministic evaluation of the design basis substantial core melt accident described in 10 CFR 50.34(a)(1)(ii)(D). RG 1.183 describes the staff guidance to address GDC 19 and Chapter 15 radiological accident analysis (see last observation on Section 15.7). Any deviations from this guidance would require extensive staff evaluation during the CP application review process. The regulations require that applicants demonstrate that the facility design can accommodate a substantial fuel

		melt source term released into an intact containment without exceeding applicable regulatory requirements. While other regulations serve to reduce the probability of a fuel melt condition, regulations requiring that the plant be able to accommodate a fuel melt source term ensures the radiological integrity of the plant design. The PSAR will need to include an evaluation of this source term or a justification for not including it. **(This observation is noted in body of the letter as needing continued engagement between TVA and NRC prior to submittal of the CP application).**
Section 3.1, PDC 26/27	GDC 27	It's not clear what is unique about the BWRX-300 regarding reactivity control for accidents when compared to other LWRs, or why there is a need for a PDC. While a PDC 27 was proposed in NEDC-33912, this should not be interpreted to imply staff determined one is actually needed.
Section 3.1, PDC 26/27	GDC 26	Proposed PDC 26/27 is not consistent with the purpose of GDC 26 to require diverse means of reactivity control. The staff's safety evaluation for NEDC-33912 clarifies that diverse motive force is not acceptable for satisfying the diverse and independent requirements of GDC 26. Information provided in subsequent RAIs form the basis of the staff's conclusions. TVA appears to be deviating from the approach described in NEDC-33912. RG 1.232, page A-13 also defines what is meant by independent and diverse with respect to GDC 26. **(This observation is noted in body of the letter as needing continued engagement between TVA and NRC prior to submittal of the CP application).**