

6 ENGINEERED SAFETY FEATURES

The staff reviewed Chapter 6 of the TVA CPA PSAR, as supplemented, against applicable regulatory requirements using regulatory guidance and standards to assess the sufficiency of the preliminary information on Engineered Safety Features for the issuance of a CP in accordance with Title 10, *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." As part of this review, the staff evaluated information on Engineered Safety Features, with special attention given to design and operating characteristics, unusual or novel design features, and principal safety considerations. The staff evaluated the preliminary design of the engineered safety features (ESFs) to ensure the design criteria, design bases, and information relative to construction are sufficient to provide reasonable assurance that the final Clinch River Nuclear Unit 1 (CRN-1) design will conform to the design basis.

The staff also reviewed the following licensing topical reports that supplement Chapter 6:

- NEDC-33911P-A, "BWRX-300 Containment Performance," GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, January 2022.
- NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," GE Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022.

The staff's reviews and evaluations for areas relevant to PSAR Chapter 6, including regulations and guidance used, a summary of the application information reviewed, and evaluation findings and conclusions, are discussed in the SER sections below for each specific review area.

6.1 Engineered Safety Features Materials

6.1.1 **Metallic Materials**

6.1.1.1 *Introduction*

The TVA CPA PSAR, Revision 1, as supplemented by letter dated March 2, 2026 (ML26061A096), described selection, fabrication, and compatibility of materials for the ESFs that consists of the primary containment system (PCS) that includes the passive containment cooling system (PCCS), steel-plate composite containment vessel (SCCV), containment isolation valves and the reactor isolation valves. The isolation condenser system (ICS) is also an ESF, but because it is part of the reactor coolant pressure boundary (RCPB), the material and controls for the ICS are discussed in Section 5.2.3. The PCS materials are compatible with the dry containment environment and are exposed only to high-quality water during normal operation. The containment closure head forms part of the reactor cavity pool and separates the pool water from containment atmosphere during normal operations. The PCS components do not come into contact with reactor coolant unless there is a postulated event.

6.1.1.1.1 *Regulatory Evaluation*

The staff based its review on the relevant requirements of the following general design criteria (GDC), regulations, and guidance:

- GDC 1, "Quality standards and records," and 10 CFR 50.55a(a)(1) require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and

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tested to quality standards commensurate with the importance of the safety functions they perform.

- GDC 4, “Environmental and dynamic effects design bases,” requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- GDC 14, “Reactor coolant pressure boundary,” requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- GDC 31, “Fracture prevention of reactor coolant pressure boundary,” requires that the design of the RCPB include sufficient margin to ensure that, when stressed under operating, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner, and the probability of rapidly propagating fracture will be minimized.
- GDC 35, “Emergency core cooling,” requires a system to provide abundant emergency core cooling. GDC 35 also requires that, during activation of the system, clad metal-water reaction will be limited to negligible amounts.
- GDC 41, “Containment atmosphere cleanup,” requires that the design provide containment atmosphere cleanup systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment. The staff limited its review of the ESF structural materials to ensure that they meet the requirements of GDC 41 with respect to corrosion rates related to hydrogen generation in postaccident conditions.
- Appendix B to 10 CFR Part 50 mandates that applicants establish quality assurance (QA) requirements for the design, construction, and prevention or mitigation of the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Compliance with 10 CFR Part 50, Appendix B is addressed in Chapter 17.
- 10 CFR 50.55a – Codes and Standards: Compliance with 10 CFR 50.55a with regard to the PCCS is addressed in Subsection 3.8.1.2 and Section 6.2. Chapter 5 addresses compliance with regard to the RCPB.

Regulatory Guidance (RG)

- RG 1.28, *Quality Assurance Program Criteria*, The quality assurance program for the BWRX-300 is described in Chapter 17.
- RG 1.31, *Control of Ferrite Content in Stainless Steel Weld Metal*, The delta ferrite content of weld filler material is controlled in accordance with the guidance of RG 1.31.
- RG 1.36, *Nonmetallic Thermal Insulation for Austenitic Stainless Steel*, Nonmetallic thermal insulation used on fluid systems performing Safety Category 1 functions is selected to minimize contamination that could promote stress corrosion cracking of the stainless-steel surface it contacts.
- RG 1.44, *Control of the Processing and Use of Stainless Steel*, The control of sensitization and the prevention of stress corrosion of stainless steels for the PCS components conforms to the guidance of RG 1.44.
- RG 1.50, *Control of Preheat Temperature for Welding of Low-Alloy Steel*, The control of preheat temperature for welding low-alloy steel for PCS components conforms to the guidance of RG 1.50.

- RG 1.54, *Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants*, Revision 3.
- Generic Letter (GL) 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping."
- NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

As described below, the staff reviewed the ESF materials in accordance with regulatory guidance and standards to meet the regulations.

6.1.1.1.2 *Technical Evaluation*

The staff reviewed PSAR Section 6.1 of the TVA CPA PSAR, Revision 0, as supplemented by letters dated January 7, 2026 (ML26007A164), and March 2, 2026 (ML26061A096), against applicable regulatory requirements using information from RGs and standards to assess the sufficiency of the preliminary information on the materials for ESFs of the BWRX300 design for the issuance of a CP in accordance with 10 CFR Part 50.

6.1.1.1.3 *Materials and Fabrication*

To meet the requirements of GDC 1 and 10 CFR 50.55a to ensure that plant SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function they perform, the applicant must identify codes and standards and maintain records. Selection of the materials specified for use in these systems must be in accordance with the applicable provisions of Section III, Divisions 1 or 2, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, or RG 1.84, *Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Revision 40*. Section III references applicable portions of ASME Code, Section II, Parts A, B, C, and D.

Table 6.1-1 in the PSAR lists the ASME Code classification and material specifications of components of ESF systems. The staff reviewed the material specifications listed in Table 6.1-1 and verified that the materials are acceptable for use in the CRN-1 design and meet the requirements of ASME Code, Section III, or the guidance of RG 1.84.

The design, fabrication, inspection and testing requirements for the ESF components, including fracture toughness requirements for all ferritic ESF materials comply with the appropriate Section III class shown in PSAR, Section 6.1 and Table 6.1-1. The preservice inspection and in-service inspection is provided in PSAR Section 6.6 for ESF system Class 2 and Class 3 as designated in PSAR Table 6.1-1.

The staff asked the applicant to include weld filler metal specifications in Table 6.1-1 (audit question A-6.1-1). In response, the applicant provided filler metal specifications and classifications for weld filler metal used in the ESF systems in Table 6.1-2. The filler metal in Table 6.1-2 is in ASME Code, Section II, Part C, "Specifications for Welding Rods, Electrodes, and Filler Metals," and therefore are acceptable.

The staff finds that the ESF materials conform to ASME Code, Section III, and RG 1.84 and that the ESF materials meet the requirements of GDC 1 and 10 CFR 50.55a.

6.1.1.1.4 *Austenitic Stainless Steel*

The CRN-1 design must meet the requirements of:

- (1) GDC 4, relative to compatibility of components with their environmental conditions
- (2) GDC 14, with respect to fabrication and testing of the RCPB so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture
- (3) QA requirements of 10 CFR Part 50, Appendix B

Designs may meet these requirements by following the guidance of Generic Letter (GL) 88-01; "NRC Position on IGSCC in BWR Austenitic Stainless-Steel Piping," NUREG-0313, Revision 2; and RGs 1.28, 1.31, and 1.44. Designs must also provide controls over the use of cold-worked austenitic stainless steels.

For stainless-steel components in the ESF systems, the PSAR Section 6.1.1.3 specifies that fabrication and processing of austenitic stainless steels will be in accordance with regulatory guidance in RGs 1.31, and 1.44. In addition, PSAR Section 6.1.4 specifies that the design complies with RGs 1.28 and 1.36, as well as GL 88-01. In addition, the base material in Table 6.1-1 and weld filler metal in Table 6.1-2 specify low-carbon stainless steel. As modified in response to Audit Question A-6.1-2, PSAR Section 6.1.1.3 describes controlling cold working during fabrication in accordance with RG 1.44 by controlling bending, forming, machining grinding, and polishing to reduce the susceptibility of components to stress-corrosion cracking. The staff finds that the use of low carbon stainless steel minimizes the occurrence of stress corrosion cracking and therefore is acceptable.

6.1.1.1.5 *Ferritic Steel Welding*

To meet the requirements of GDC 1 related to general quality assurance and codes and standards, Appendix B to 10 CFR Part 50 for control of special processes, and 10 CFR 50.55a, the minimum specified preheat must meet RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," unless an alternative procedure is justified. In addition, moisture control on low-hydrogen welding materials must conform to the requirements of ASME Code, Section III.

PSAR Section 6.1.1.1 and Table 6.1-1 specifies that the design, fabrication, installation, and inspection of PCS components are based on Subsection NCA and NCD of Section III to the ASME Code. Therefore, the applicant will be controlling the storage and handling of filler metal to minimize the absorption of moisture by welding electrodes and fluxes, and therefore the staff finds moisture control of welding materials acceptable.

Table 6.1-2 identifies the applicable weld material for carbon steel and low alloy steel. In addition, the guidelines of RG 1.50 will be followed for preheat and interpass temperature. The staff finds that the applicant's compliance with the guidelines in RG 1.50 is acceptable, since it provides reasonable assurance that delayed hydrogen cracking will not occur.

6.1.1.1.6 *Composition and Compatibility of ESF Fluids*

PSAR Table 1.9-6 states that the composition and compatibility of the ESF fluids conforms to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" March 2007 (hereafter referred to as the SRP), Section 6.1.1. To meet the requirements of GDCs 4, 14, and 41, the design should control the chemistry of the water used in the ESFs to ensure against stress corrosion cracking (SCC) in unstabilized

stainless-steel components. Because the chemistry control requirements of the ESF fluid will conform to SRP 6.1.1, the staff finds that the ESF fluid is compatible with the ESF materials.

PSAR Section 6.1.1.2 refers to PSAR Section 3.8.1.4.8 for additional information on corrosion prevention. PSAR Section 3.8.1.4.8 requires corrosion protection in accordance with Section 5.15 of NEDC-33926P, by using one or a combination of the following approaches to meet the design life of the components:

- corrosion tolerance by adding a sacrificial thickness
- protective paint system suitable to the surrounding environment
- membrane coating system
- impressed current cathodic protection

The staff considers the applicant's corrosion protection acceptable, given that they are general methods for controlling corrosion on carbon and low alloy steels .

6.1.1.1.7 Component and System Cleaning

The staff reviewed the ESF structural materials to ensure that the requirements of Appendix B to 10 CFR Part 50 were met, as they relate to the establishment of measures to control the cleaning of material and equipment. The controls established for cleaning of material and equipment must be performed in accordance with work and inspection instructions to prevent damage or deterioration.

PSAR Section 6.1.4, specifies that the guidance in RG 1.28 is applicable to the ESF materials. RG 1.28 endorsed the use of ASME NQA-1-2015, which has strict process controls for cleaning and protection against contamination of materials during all stages of component manufacture and installation. For example, tools used in abrasive work on austenitic stainless steel, such as grinding, should not contain and should not have been contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or SCC. Therefore, the staff finds the applicant's cleaning and cleanliness controls for ESF components acceptable and consistent with SRP Section 6.1.1.

6.1.1.1.8 Thermal Insulation

PSAR Section 6.1.4 states that RG 1.36 is applicable to the ESF components as it relates to selection, procurement, testing, and storage of nonmetallic thermal insulation materials used on ESF systems.

To meet the requirements of GDCs 1, 14, and 31, ESF systems should be designed, fabricated, erected, and tested such that there is an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The levels of leachable contaminants in nonmetallic insulation materials that come into contact with 300 Series austenitic stainless steels used in fluid systems important to safety should be under careful control so as not to promote SCC. In particular, the leachable chlorides and fluorides should be held to the lowest levels practical. The staff's position is that following the guidance in RG 1.36 is an acceptable method to control leachable contaminants in nonmetallic insulation materials. The applicant has stated that it will follow the guidance in RG 1.36, and the staff finds this acceptable as it will meet the requirements of GDCs 1, 14, and 31.

6.1.1.1.9 *Organic Materials*

The staff reviewed the protective coating systems (paints) and organic materials in accordance with SRP Section 6.1.2, Revision 3. Staff acceptance is based on meeting the requirements of Appendix B to 10 CFR Part 50, as it relates to the QA requirements for the design, fabrication, and construction of safety-related SSCs. To meet the requirements of Appendix B to 10 CFR Part 50, the applicant should specify that the coating systems and their applications will follow the guidance of RG 1.54, Revision 3. This RG references the standards of ASTM D5144-00, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants."

RG 1.54, Revision 3, provides guidance on practices and programs that are acceptable to the staff for the selection, application, qualification, inspection, and maintenance of protective coatings used in nuclear power plants. PSAR Section 6.1.2 specifies that protective coatings are qualified and capable of surviving a design basis accident (DBA) without adversely affecting PCS components needed to mitigate the accident in accordance with the guidance of RG 1.54, Revision 3. Additional information on RG 1.54, Revision 1, conformance is provided in PSAR Sections 3.8.1.4.8 (SCCV diaphragm plates steel-plate composite modules) and 3.8.2.4 (design of the containment closure head and the Class MC steel components). The staff finds the organic coatings to be acceptable because coatings inside containment will meet the requirements of RG 1.54, Revision 3, and are qualified using standard ASTM tests.

6.1.1.1.10 *Conclusion*

The applicant provided sufficient information on the selection of materials, fabrication processes, compatibility of materials, and cleaning and cleanliness controls that are acceptable in accordance with guidance in RG 1.70 because they satisfy regulatory requirements or positions described above, or because they have been demonstrated to be acceptable based on appropriate materials selections and acceptable operating experience.

Based on the above, the staff concludes that the ESF materials are acceptable and can meet the requirements of GDCs 1, 4, 14, 31, 35, and 41; Appendix B to 10 CFR Part 50; and 10 CFR 50.55a. Therefore, the NRC staff finds that the information in PSAR Section 6.1 is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a) and 10 CFR 50.35, as applicable.

6.2 Containment System

6.2.1 **Containment Functional Design**

The BWRX-300 design uses passive systems as the ESFs that do not depend on operator actions or external power, to fulfill the safety functions for at least 72 hours after a DBA. The BWRX-300 containment (i.e., the PCS), is an ESF that provides a physical barrier against radiological releases to the environment. Containment heat removal during accident conditions is provided by the PCCS. Containment isolation valves, piping, and penetrations are part of the containment boundary. PSAR Sections 6.2.1 and 6.2.2 describe the PCS and PCCS, respectively. The following SER Sections 6.2.1 and 6.2.2 are drafted for the applicant's PSAR following the acceptance criteria of the respective Chapter 6 SRP sub-sections in NUREG-800.

PSAR Table 1.9-6 tabulates the conformance with NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, for PSAR Chapter 6, while PSAR Table 3.1-1 provides additional information about the compliance with various General Design Criteria/Principal Design Criteria involved.

6.2.1.1 Primary Containment System

The BWRX-300 PCS is a dry and essentially leak-tight containment which is inerted with nitrogen using the containment inerting system (CIS), as discussed in Subsection 9.3.6. The PCS, which is mostly below-grade and housed within the Reactor Building, encloses the reactor pressure vessel (RPV) and does not incorporate a pressure suppression pool or a secondary containment. Table 6.2-1 provides key PCS design parameters. Figure 3.8-1 and Figure 3.8-2 show the structural boundary of the BWRX-300 containment, containment internal structures, and Reactor Building. Containment heat removal during normal operation is provided by the containment cooling system, as discussed in Subsection 9.4.7. During postulated accidents, containment heat is removed by the PCCS that is a part of the PCS described in PSAR Section 6.2.2. The PCS is designed to maintain its integrity for at least 72 hours without operator action.

PSAR Chapter 6 provides descriptions of the containment functional design and heat removal systems. However, the containment analysis descriptions and results to support the safety evaluation of the BWRX-300 containment safety design are included in PSAR Chapter 15. Chapter 15, in general, provides the plant response to design-basis events, including the ones that are relevant to the PSAR Chapter 6 containment safety analyses. The applicant's PSAR references GEH topical report NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," Revision 3 (Reference 6.2-2) where the BWRX-300 containment evaluation model was approved. The approval of the model in the TR report included two limitations and conditions (L&C), L&C#3 and L&C#4, associated with PCCS performance that are also addressed in PSAR Section 15.5.4.5, along with the remaining two limitations and conditions, L&C#1 and L&C#2, that are associated with ICS performance. The containment safety under DBAs is reviewed under the NRC regulations; GDC 16, GDC 50, and GDC 38, and acceptance criteria that are documented in SRP Sections 6.2.1 and 6.2.2. Even though the Chapter 15 and Chapter 6 safety analyses have different figures of merit and conservative biases, the following large-break loss of coolant accident (LBLOCA) and small-break loss of coolant accident (SBLOCA) containment thermal-hydraulic response results presented in the following Chapter 15 figures for various pipe breaks pertain to the Chapter 6 containment safety analyses and figures of merit: Figures 15.5-102, 15.5-104, 15.5-105, 15.5-109, 15.5-110, 15.5-111, 15.5-115, 15.5-116, 15.5-117, 15.5-118, 15.5-164, 15.5-165, and 15.5-166. The results presented in these figures are key to some of the safety findings made in Sections 6.2.1 and 6.2.2 of this SER.

6.2.1.1.1 Regulatory Evaluation

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are given in SRP Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containment," Revision 3, and are summarized below:

- GDC 13, "Instrumentation and Control," as it relates to instrumentation and control, requires instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to assure adequate safety.

- GDC 16, “Containment Design,” as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Because the primary reactor containment is the final barrier of the defense-in-depth concept to protect against the uncontrolled release of radioactivity to the environment, preserving containment integrity under the dynamic conditions imposed by postulated loss of coolant accidents is essential.
- GDC 38, “Containment Heat Removal,” as it relates to the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain the reduced conditions at acceptably low levels.
- GDC 50, “Containment Design Basis,” as it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any LOCA without exceeding the design leakage rate and with sufficient margin.
- GDC 64, “Monitoring Radioactivity Releases,” as it relates to requiring that means be provided for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.
- 10 CFR 50.34(f)(3)(v)(A)(1), as it relates to containment integrity being maintained during an accident that releases hydrogen generated from a 100 percent fuel clad metal-water reaction accompanied by hydrogen burning.

The guidance in SRP Section 6.2.1.1.A lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

6.2.1.1.2 *Technical Evaluation*

Specific SRP acceptance criteria to meet the relevant requirements of the NRC regulations identified above for the “PWR Dry Containments, Including Subatmospheric Containments,” are evaluated as follows for the review described in this SER section.

GDC 16 requires reactor containment to be designed as a leak-tight barrier that would withstand the most extreme design-basis accident conditions and preclude the uncontrolled release of radioactivity to the environment, while GDC 50 requires the containment structure and associated heat removal system to be designed to accommodate any postulated accident conditions, without exceeding the containment design conditions that are important to safety (e.g., the design leakage rate, with sufficient margin). In this regard, SRP Acceptance Criterion #1 ensures the containment integrity based on the deterministic safety analyses of the most severe DBAs for peak pressure and temperature conditions for the containment. Per Acceptance Criterion#1, to satisfy the requirements of GDC 16 and 50 regarding sufficient design margin, for plants at the CP stage of review, the containment design pressure should provide at least a 10-percent margin above the accepted peak calculated containment pressure following a LOCA or a steam or feedwater (FW) line break.

PSAR Section 3.1.2.7 presents an evaluation of the SCCV enclosing the RPV design against GDC 16. It describes that the temperature and pressure conditions inside the SCCV are controlled and maintained below acceptance criteria following an accident for at least 72 hours by RPV decay heat removal using the ICS and condensation on the SCCV walls and with containment heat removal using the PCCS. PSAR Chapter 15 presents the containment temperature and pressure response analysis results and evaluation for the limiting DBAs for

the referenced BWRX-300 design against GDC 16, GDC 50, and GDC 38 of Appendix A to 10 CFR Part 50. The PSAR follows the methodology for evaluating the containment structure and heat removal systems to withstand the effect of potential accidents, including margin, using the GEH topical report NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," Revision 3 (Reference 6.2-2) which is incorporated by reference in the PSAR (Reference 1.6-2). NEDC-33922P-A was approved by NRC for the analysis method used for the thermal hydraulics performance of the containment design against the acceptance criteria listed in NEDC-33911P-A, "BWRX-300 Containment Performance," that also is referenced in PSAR Chapters 6 and 15 (Reference 6.2-3, Reference 15.5-20). NEDC-33911P-A, Revision 3, was approved by the NRC for the design requirements, analytical methods, acceptance criteria, and regulatory basis for the BWRX-300 containment performance design functions, as used in the PSAR. This licensing topical report (LTR) contains information that needs to be part of the PSAR to make a regulatory/safety finding. Therefore, the staff issued Audit Question A-6.2-15 to request the applicant to add the relevant information from the LTR NEDC-33911P-A in the PSAR or incorporate by reference in the PSAR the part needed from the LTR. In response, the applicant identified and added the following relevant information from LTR NEDC-33911P-A, Revision 3, "BWRX-300 Containment Performance" to the PSAR and provided the needed PSAR mark-ups.

- Technical descriptions of the BWRX-300 containment, PCCS, ICS, and the containment isolation valve (CIV) design features and design functions, acceptance criteria, regulatory bases provided in Section 2.0 of NEDC-33911P-A are updated by the information in PSAR Section 3.8 and Chapter 6.
- BWRX-300 containment performance acceptance criteria provided in Section 4.0 of NEDC-33911P-A are added to PSAR Chapter 6.
- Regulatory evaluations provided in Section 5.0 of NEDC-33911P-A are updated in various PSAR sections as shown on the submitted PSAR markup. The applicant also provided a crosswalk of the LTR regulatory evaluation and the corresponding PSAR sections (including updates in PSAR through Supplement 5 and the attached markup).

All the above information along with the PSAR mark-ups was also submitted on the TVA CPA docket (ML26035A332). Therefore, the staff concludes that the applicant has addressed the staff concerns about the potential inconsistencies between the regulatory bases approved as a part of LTR NEDC-33911P-A, Revision 3, "BWRX-300 Containment Performance" and the regulations followed by the PSAR.

PSAR Table 15.7-3 summarizes the key results for DBA for LOCAs. Based on the containment thermal-hydraulic response results for the PSAR, an LBLOCA resulting from the large main steam (MS) pipe break inside the containment is identified as the limiting BWRX-300 DBA event for both containment peak pressure and temperature cases. Containment peak pressure and temperature are limited by the heat removal. Decay heat is rejected to the IC pools directly from the RPV by the ICS, as well as through containment heat removal system using the PCCS by condensation and natural circulation, and condensation and convection to the containment walls. The PCCS does not have a high duty function for long-term containment cooling as most of the decay heat is removed by the ICS. Three independent trains of PCCS are included in the design to provide a high degree of reliability. The PCCS does not require electrical power to operate, and the ICS operation only requires one-time actuation using onsite Safety Category 1 (SC1) battery-backed DC power. Integral reactor isolation valves (RIVs) that are directly flanged to the RPV nozzles on the reactor vessel allow large break LOCAs to be quickly isolated, thereby preserving the fluid inventory in the RPV for core cooling. Small breaks less than or equal to 19 millimeters (0.75 inches) do not need to be isolated. SRP Figure 15.5-104 shows

that both the rated initial conditions and hot shutdown initial conditions are analyzed for the conservative case. As shown in Figure 15.5-104, the staff agrees that the rated initial condition is more limiting for containment peak pressure, as hot shutdown conditions would result in higher liquid break flow and lower enthalpy. Table 15.7-3 documents the calculated peak containment pressure of 423 kPa (321.7 kPaG) and shell temperature of 134°C for the limiting containment DBA (i.e., the large MS pipe break). Table 6.2-1 that documents the primary containment system key design parameters tabulates 60.0 psig (413.7 kPaG) as the containment design internal pressure, and 165.5°C as the containment design temperature. This shows that the calculated peak containment pressure (321.7 kPaG) is less than the design pressure (413.7 kPaG) with more than 20-percent margin, and the calculated peak containment shell temperature (134°C) also is significantly lower than the containment design temperature (165.5°C). Figure 15.5-105 shows the containment air and shell temperatures for the conservative case of the limiting large MS pipe break event. The staff finds it acceptable that the containment air temperatures, as shown in Figure 15.5-105, do exceed the containment design temperature 165.5 °C, as the maximum inner and outer shell surface temperatures (134°C) still remain below the containment design temperature.

Figures 15.5-165 and 15.5-166 show the pressure and temperature response for the conservative case of the FW pipe break. As in the MS pipe break cases, containment pressure on Figure 15.5-165 and temperature on Figure 15.5-166 are calculated for break locations maximizing pressure and temperature. Figure 15.5-165 shows the containment pressure for normal FW temperature and reduced FW temperature. Normal FW temperature results in a higher containment pressure. The pressure and temperature response of the large FW pipe break for both cases is similar but is bounded by the peak pressure resulting from an MS pipe break. Figures 15.5-110 and 15.5-111 show the pressure and temperature response for the conservative case of the limiting small steam pipe break event with loss-of-preferred power (LOPP). Containment pressure and temperature response shown on Figure 15.5-110 and Figure 15.5-111 for a small steam pipe break is similar to the containment response for the conservative case of the small liquid pipe break event, as shown in Figure 15.5-117 and Figure 15.5-118. Key results for all analyzed breaks are summarized in Table 15.7-3 and show that the peak containment pressure and temperature for the limiting small steam and liquid breaks, and FW pipe break are bounded by the limiting LBLOCA resulting from the large MS pipe break. Figures 15.5-110 and 15.5-117 show that containment pressure for both steam and liquid SBLOCAs remains below 262 kPa in the long term, which is 50 percent of the peak accident pressure for the limiting LBLOCA.

Per SRP Acceptance Criterion#2, GDC 38 requires the establishment of a containment heat removal system that will rapidly reduce containment pressure and temperature following any LOCA and will maintain them at acceptably low levels, thereby protecting the safety function of the containment as an engineered safety feature. Meeting GDC 38 helps ensure that the containment can fulfill its function while maintaining its integrity as the final barrier against the release of radioactivity to the environment, by minimizing the duration and intensity of the pressure increase following a LOCA. Per Acceptance Criterion#2, to satisfy the requirements of GDC 38 to rapidly reduce the containment pressure, the calculated containment pressure should be reduced to less than 50 percent of the peak calculated pressure for the design basis LOCA within 24 hours after the postulated accident. Table 1.9-6 documents the TVA compliance with GDC 38 regarding rapid reduction of containment pressure. Table 15.3-2, Design Basis Accident Deterministic Safety Analysis Acceptance Criteria, also explicitly states that the containment capability will be retained to reduce the containment pressure and temperature following a DBA to minimize the release of fission products to the environment and to preserve containment integrity and leak tightness. PSAR Section 15.5.4.5.1 describes that as

the peak accident pressure for the bounding conservative case of the MS pipe break inside containment is approximately 322 kPaG (423 kPa), the half of the peak pressure is approximately 161 kPaG (262 kPa). The staff noted that Figure 15.5-104 shows that the containment is depressurized rapidly, and the calculated containment pressure reduces to less than 50 percent of the calculated peak pressure for the most limiting LOCA within a few hours. This meets the acceptance criterion for the containment response to pipe breaks that the containment pressure should be reduced to less than half of the peak pressure in 24 hours. Additional information discussing the compliance approach to GDC 38 is described in the applicable PSAR sections listed in Table 3.1-1. The staff also conducted an audit of the engineering calculations to make the safety finding regarding the requirements of 10 CFR Part 50, Appendix A, GDC 38. In this regard, several Audit Questions (A-6.2-10, A-6.2-11, A-6.2-12, A-6.2-14, and A-6.2-17) were issued. Based on the staff audit and the resolution of the audit questions, the staff concludes that the requirements of 10 CFR Part 50, Appendix A, GDC 38 have been met in the PSAR for the issuance of a CP.

Per SRP Acceptance Criterion#4, to satisfy the requirements of GDCs 38 and 50 with respect to the containment heat removal capability and design margin, the LOCA analysis should be based on the assumption of loss of offsite power and the most severe single failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure. Table 1.9-6 documents the TVA compliance with GDC 38 and GDC 50 regarding containment heat removal and design margin for LOCA analysis. The Figures 15.5-110 and 15.5-111 results described above for the small steam pipe break event were obtained with LOPP. The staff accepts LOPP as the same as a loss of offsite power for the BWRX-300 design, as it can be used interchangeably to represent a loss of the switchyard as a power source to the plant. PSAR Section 6.2.2 describes that the PCCS can achieve its function with a single failure and does not require alternative current (AC) or direct current (DC) power to function and can meet heat removal requirements in the event of a single failure. A staff audit of BWRX-300 GOTHIC containment analysis details confirmed that all Chapter 15 containment analyses are based on two PCCS trains being functional out of the three available in the design. Therefore, the PCCS can achieve its safety function with a single failure of one PCCS train and the containment heat removal system redundancy and power source consistent with ESF design conform to the loss of offsite power and the most severe single failure requirements of GDCs 38 and 50.

As noted in PSAR Table 1.9-6, TVA stated conformance with the applicable acceptance criteria contained in NUREG-0800, Sections 6.2.1.1.A and 6.2.1.3. The staff has reviewed the related information provided in PSAR Sections 6.2.1, 6.2.2, and 15.5.4.5. To conclude that SRP Acceptance Criteria#1, #2, and #4 have been met, the staff audited the related engineering calculations to ensure the compliance Appendix A of with 10 CFR Part 50, GDC 16, GDC 50, and GDC 38 regarding containment design pressure and shell temperature margins and the application of the containment evaluation methodology as approved in NEDC-33922P-A that is incorporated by reference in the PSAR. In this regard, the staff needed to evaluate the applicant's GOTHIC and Transient Reactor Analysis Code General Electric (TRACG) models as used in the containment DBA safety analyses and results presented in PSAR Section 15.5.4.5, and GEH report, NEDC-34349P, "BWRX-300 DNNP Joint Report on GEH's Containment Evaluation (CE) Method: Implementation of Licensing Considerations," as provided by the applicant for audit. The staff needed to assess the DBA models and input parameters used for the PSAR mass and energy (M&E) release and containment response analyses to ensure that the analysis is acceptably conservative and satisfies the requirements of GDCs 16 and 50 regarding sufficient design margin for calculated pressure and temperature conditions.

Specifically, the staff needed to evaluate whether the safety-significant phenomena were conservatively modeled, consistent with the guidance in SRP Sections 6.2.1.1.A and 6.2.1.3, over the applicable range of accident conditions, and whether the four L&Cs imposed on the LTR NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," were appropriately addressed. Therefore, the staff issued Audit Question A-6.2-9 to request an audit of the QA-checked engineering calculation notebook[s] that provides details of the PSAR containment design parameters, initial/boundary conditions, and modeling assumptions, used in the GOTHIC and TRACG input model decks nodalized and biased for the Chapter 6 peak containment pressure/temperature DBA analyses. The documentation needed to cover the limiting containment DBAs for large and small breaks inside the containment, for the applicable LOCA, MS line break, and main FW line break cases. The staff used the information to evaluate the application of the NEDC-33922P-A CE methodology with the modified PCCS design, and addressing the four L&Cs for the CP application, especially by verifying the following information provided in the PSAR to demonstrate conformance with the applicable acceptance criteria contained in NUREG-0800 Sections 6.2.1.1.A and 6.2.1.3.

The staff conducted an audit of the engineering calculations and results in response to Audit Question 6.2-9 to verify the conservatism and initial/boundary conditions, in regard to GDC 16, GDC 50, and GDC 38 acceptance criteria. The staff looked into the PDF and Excel files to verify the conservatism and initial/boundary conditions to finalize the safety findings. The staff also compared the containment DBA results presented in PSAR Chapter 15 with the ones demonstrated in the Topical Report (LTR) NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," Revision 3 (Reference 6.2-2), and observed some major differences in results and trends that the audit could not explain. The staff observed that peak containment pressure (PCP) and temperature values, trends, and timings for the conservative case of the limiting MS large break LOCA were almost identical for the PSAR (Figures 15.5-104, 15.5-105) and LTR (Figures 6-26, 6-27). However, the FW pipe break results were different in the PSAR and LTR. NEDC-33922P-A presented 266.8 kPa PCP (LTR Figure 6-9) for the FW pipe break base case but reported no results for the conservative case. While the FW pipe break base case PCP was 93.4 kPa less than the MS pipe break base case PCP (LTR Figure 6-4, 360.2 kPa), the PSAR's 407 kPa PCP for FW pipe break conservative case is only 16 kPa less than the MS pipe break conservative case PCP (423 kPa) and 140.2 kPa higher than the FW pipe break base case PCP from the LTR. These variations suggested a significant change in the FW pipe break response in the PSAR, as compared to those in the LTR. There also were significant differences between the PSAR (Figures 15.5-110, 15.5-111, 15.5-117, 15.5-118) and LTR (Figures 6-31, 6-34, 6-39, 6-41) results for peak containment pressure and temperature values, trends, and timings for the conservative cases of the limiting small steam and liquid break LOCAs, as described below.

- The PSAR steam and liquid SBLOCA peak containment pressures (191 kPa) are much smaller than the LTR peak values (243.2 and 253.9 kPa) and occur much later ($\sim 2.3 \times 10^5$ sec) than in the LTR ($\sim 3 \times 10^3$ sec). After the containment pressure peaked much earlier in the LTR, it significantly dropped until it equalized with the RPV pressure around ~ 200 kPa at around $1-2 \times 10^5$ sec for both steam and liquid SBLOCAs. On the contrary, the containment pressure in the PSAR does not peak and, rather, keeps increasing monotonically until it reaches its maximum value when it equalizes with the RPV pressure.
- The PSAR and LTR peak shell temperature trends and timings are similar for both steam and liquid SBLOCAs, but the peak shell temperature values are significantly different, even though they occur around the same time. The PSAR peak shell temperature values are

15.2°C lower for steam break (125°C for PSAR vs. 140.2°C for LTR), and 13.5°C lower for liquid break (110°C for PSAR vs. 123.5°C for LTR).

The staff was unable to understand these differences while the same PIRT and M&E release are applicable in PSAR as in LTR. Because the staff could not audit the TRACG or GOTHIC decks or run confirmatory calculations, it could not establish whether any changes in the containment design and/or the CE methodology were causing these differences. Therefore, the staff issued Audit Question A-6.2-10 to request the applicant to provide justification for the safety-significant differences between the LOCA response results and trends in the PSAR and the Topical Report NEDC-33922P-A, "BWRX-300 Containment Evaluation Method." The staff needed to understand these differences before making the safety findings for several acceptance criteria in SRP Sections 6.2.1.1A, 6.2.1.3, and 6.2.2 as well as for addressing the four limitations and conditions on NEDC-33922P-A in PSAR Section 15.5.4.5 that are primarily applicable to un-isolated small breaks. The TVA response provided the following key information regarding the staff concerns.

- The applicant addressed the staff concern that the PCP and temperature values, trends, and timings for the conservative case of the limiting MS pipe break LBLOCA were about the same for the PSAR and LTR, but the FW pipe break LBLOCA results were significantly different in the PSAR and LTR. The TVA response explained that the differences in the FW pipe LBLOCA in the PSAR and LTR are caused by having more water in a larger FW piping volume assumed in the PSAR case, which results in more flashing of water through the FW pipe break and, thus, a higher PCP. The staff agrees that the LTR-to-PSAR change due to conservatively assuming higher water volume could lead to a higher PCP (407 kPa) for the FW pipe break that would be conservative and would still not challenge the limiting large MS pipe break conservative case PCP (423 kPa). The staff also agrees that assuming feed water to be saturated is conservative as it is likely to be subcooled. The staff concludes that changes in the FW pipe break results are not indicative of a non-conservative variation from the containment methodology and are, therefore, acceptable.
- The applicant also addressed the significant differences noticed by the staff between the PSAR and LTR results for PCP and temperature values, trends, and timings for the conservative cases of the limiting small steam and liquid break LOCAs. The applicant's response explained that the difference in the small break responses is due to different small break sizes used in the PSAR and LTR. While the LTR demonstrated that the containment pressure and temperature could be maintained below the design pressure and temperature and meet the containment acceptance criteria for a 25 mm small break, the actual un-isolated instrument line breaks of BWRX-300 are limited to an area of 19 mm equivalent diameters in the PSAR. The staff accepts this explanation because it is consistent with the upper limit of the small break size used in the PSAR Chapter 6 and Chapter 15.

For the purpose of issuing a CP, the staff accepts the qualitative explanations provided by the applicant to address the staff concerns about the PSAR-LTR LOCA differences. However, the staff will perform an audit of the underlying calculations for the final design as a part of the final safety analysis report (FSAR) review at the operating license (OL) stage.

Per SRP Acceptance Criterion#3, to satisfy the requirement of GDC 38 for sub-atmospheric containments, the containment pressure should be rapidly reduced to below atmospheric pressure within 1 hour after the postulated accident, and the sub-atmospheric condition should be maintained for at least 30 days. Even though this criterion would only be applicable to sub-atmospheric containments and the BWRX-300 containment is not sub-atmospheric, Table

1.9-6 documents its conformance for the TVA containment design. The staff concludes that Table 1.9-6 needs to document that the corresponding GDC 38 requirement for rapid reduction of containment pressure for sub-atmospheric containments is not applicable to the BWRX-300 design. However, the staff does not have any safety concerns in this regard.

Per SRP Acceptance Criterion#5, the containment heat removal capability and design margin requirements of GDCs 38 and 50 need to be satisfied for the containment response analysis for postulated secondary system pipe ruptures. Even though this criterion would only be applicable to pressurized water reactors (PWRs), Table 1.9-6 documents its conformance for the TVA containment heat removal system design for the secondary system pipe ruptures, even though PSAR Section 6.2.1.4 states that the BWRX-300 design does not use a secondary system for FW or steam production. The staff concludes that Table 1.9-6 needs to document that SRP Section 6.2.1.1.A, Acceptance Criterion#5 does not apply to the BWRX-300 design. However, the staff does not have any safety concerns in this regard.

Per Acceptance Criterion#6 of SRP Section 6.2.1.1.A, to satisfy the requirements of GDCs 38 and 50 with respect to the functional capability of the containment heat removal systems and containment structure under LOCA conditions, provisions should be made to protect the containment structure against possible damage from any resulting external pressure conditions. PSAR Table 1.9-6 specifies that the referenced design conforms to the external pressure requirements of GDC 38 and GDC 50 regarding the containment heat removal and design margin. Even though Table 1.9-6 cites PSAR Section 6.2.1, the staff could not locate any related description in the PSAR. PSAR Table 6.2.1 reports a maximum negative pressure of -2 psig for the containment that is normally maintained at an inerted atmosphere at a slight positive pressure (0.25–1.30 psig). However, the PSAR provides no assurance that the containment structural design is conservative and is capable of withstanding the maximum expected external pressure. It does not describe any provisions to preclude inadvertent operation of containment heat removal systems, either. Therefore, the staff issued Audit Question A-6.2-13 to request that the applicant provide the information to support the containment external pressure design evaluation and confirm whether -2 psig is an adequate margin for the maximum expected external pressure to account for uncertainties in the analysis of the postulated external pressure event. The applicant's response provided following reasons to support the bounding external pressure of -2 psig for the BWRX-300 primary containment.

- The bounding external pressure around -2 psig is a typical value used for containments with active systems. However, unlike conventional plants, the BWRX-300 containment is passively cooled by the PCCS that is always in service but only becomes effective at removing heat in a steam environment.
- The most significant negative pressure is not expected to be caused by inadvertent actuation or accidents but, rather, by overcooling by normal containment cooling, which is a much milder transient than inadvertent actuation of an active cooling system. However, the BWRX-300 does not have active cooling systems such as containment sprays that could inadvertently actuate.

The staff agrees that as the magnitude of the bounding external pressure for BWRX-300 is expected to be less than that of the conventional plants with active cooling systems, using a preliminary -2 psig design value for the detailed design at the PSAR stage would provide reasonable assurance for the final containment external pressure design flexibility. Therefore, the staff accepts that the corresponding analysis has not been completed for the PSAR, and the final calculations will be performed after detailed design is complete. The applicant also has made a commitment that the analysis confirming maximum external pressure will be

available for the FSAR, and if there is any challenge to the final containment design integrity, the normal containment cooling and the containment purging system designs will be adjusted as needed. Based on engineering judgment, the staff concludes that applicant has addressed staff concerns about the evaluation of external pressure conditions for the GDCs 38 and 50 requirements per SRP Section 6.2.1.1.A. and has provided reasonable assurance adequate for the CP issuance.

Per SRP Acceptance Criterion#7, GDCs 13 and 64, and 10 CFR 50.34(f)(2)(xvii) have requirements for the provision of instrumentation capable of monitoring the containment atmosphere pressure and temperature and the sump water level and temperature in the post-accident environment. Because containment plays a vital safety role, appropriate instrumentation, such as for temperature and pressure measurements, must be provided so that operators can verify that containment is properly fulfilling its function. The instrumentation should have adequate range, accuracy, and response to assure that these parameters can be tracked and recorded throughout the course of an accident. SRP Section 6.2.1.3 states that the design of the PCS includes instrumentation capable of operating in the post-accident environment to monitor containment parameters, as well as main control room (MCR) monitoring of instrumentation to measure containment pressure, water level, hydrogen concentration, radiation level, and noble gas effluents. GDC 13 and GDC 64 require that instrumentation be provided to monitor all expected parameters of normal operation, anticipated operational occurrences, and accidents to assure adequate reactor safety is maintained. RG 1.97 provides specific criteria for acceptable design of containment instrumentation fulfilling GDC 13 and GDC 64. Meeting GDC 13 and GDC 64 along with the specific guidance of RG 1.97, as elaborated on by Branch Technical Position 7-10, would help ensure that containment accomplishes its mission of precluding the release of radioactivity to the environment. PSAR Section 3.1.2.4 provides the TVA CP-specific information to evaluate the GDC 13 compliance for instrumentation and control. Additional information discussing the compliance approach to GDC 13 is described in the applicable PSAR sections listed in Table 3.1-1. Likewise, SRP Section 3.1.6.5 provides the TVA CP-specific information to evaluate the GDC 64 compliance for monitoring containment atmosphere for radioactivity releases from normal operations, anticipated operational occurrences, and accidents. Additional information discussing the compliance approach to GDC 64 is described in the applicable PSAR sections listed in Table 3.1-1. Summarily, PSAR Table 1.9-6 confirms that the Chapter 6 ESFs comply with GDC 13, GDC 64, and 10 CFR 50.34(f)(2)(xvii) regarding post-accident containment monitoring. Additional information is provided in PSAR Table 1.9-21 and Sections 6.2.1. The staff concludes that enough assurance has been provided in the PSAR for CP purposes regarding the PCS instrumentation to monitor parameters important to the containment and its associated systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety.

Acceptance Criterion #8 of the SRP Section 6.2.1.1.A pertains to 10 CFR 50.46 Appendix K, I.D.2, which requires that the minimum calculated containment pressure should not be less than that used in the analysis of the emergency core cooling system (ECCS) capability. However, PSAR Table 1.9-6 documents that compliance with 10 CFR 50.46 Appendix K.I.D.2 regarding containment pressure for ECCS capability during reflood and spray cooling is not applicable to the BWRX-300 design. PSAR Section 6.2.1.5 also states that "the BWRX-300 design does not rely on containment pressure to achieve emergency core cooling, as discussed in NEDC-33911P-A." The staff has noted that large break LOCAs get isolated before containment pressure could affect the ECCS design functions, and small pipe breaks do not need to be isolated and are mitigated by the elevated placement of the small nozzles and proper sizing of the ICS and PCCS systems. As the coolant inventory available inside the RPV is demonstrated

to be sufficient and containment and RPV pressures equalize along with the downcomer and the ICS operation, adequate core cooling is maintained for both LBLOCA and SBLOCA without safety-related high-pressure injection. Therefore, the staff agrees that Acceptance Criterion #8 of the SRP Section 6.2.1.1.A about the minimum calculated containment pressure is not applicable to the BWRX-300 design that does not rely on containment pressure to achieve emergency core cooling.

SRP Acceptance Criterion#9 requires that in accordance with GDC 4, containment internal structures and system components (e.g., reactor vessel and supports) should be designed to withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment sub-compartments. PSAR Table 1.9-6 and Section 6.2.1 document that the PCS conforms to the environmental and dynamic effects design bases requirements of GDC 4, which has plant-wide applicability. PSAR Section 3.1.1.4 evaluates the referenced BWRX-300 design against GDC 4 for the environmental and dynamic effects design bases. The SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. This includes the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, the dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. Compliance regarding internal structures and component capability to withstand differential pressure loadings is further evaluated in SER Section 3.1. The staff concludes that enough information has been provided in the PSAR to meet Acceptance Criterion#9 for the CP approval purposes.

Compliance with the Three Mile Island Unit 2 related Acceptance Criterion#10, 10 CFR 50.34(f)(3)(v)(A)(1), about hydrogen control requires that the containment be designed to withstand either hydrogen burning or initiation of the post-accident inerting system, if installed, during an accident that releases hydrogen from a 100 percent fuel clad metal-water reaction. This requires evaluation of a postulated accident with hydrogen generation from a 100-percent fuel clad metal-water reaction to demonstrate that the appropriate article for service level C limits (considering pressure and dead load only), for either concrete or steel containments, from ASME Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this initial accident, the increase in pressure from hydrogen burning in containment needs to be analyzed. The containment must be designed to withstand this additional pressure to ensure that its integrity is maintained, thus precluding the release of radioactivity to the environment. PSAR Section 3.1.4.6 states that the BWRX-300 acceptance criteria in response to a LOCA is to maintain the reactor water level at or above the top of active fuel or fuel cladding temperature is maintained within normal operating temperature range, such that the performance of the ICS and RPV isolation valves is sufficient to ensure that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. PSAR Section 6.2.1.3.1 and Table 1.9-6 also make a commitment that compliance with 10 CFR 50.34(f)(3)(v)(A)(1) will be demonstrated by addressing an evaluation of 100 percent fuel clad metal-water reaction in the FSAR. The staff expects that the required demonstration of the ECCS safety function in the FSAR at the OL application stage needs to show heat removal from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

6.2.1.1.3 Conclusion

The staff performed a review of the TVA PCS information provided in the PSAR against the relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria specified in SRP Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containment," Revision 3. The staff's review of the TVA containment design compliance with the regulatory requirements led to issuing several audit questions to the applicant. Applicant responses were received, and all related audit issues (A-6.2-10, A-6.2-11, A-6.2-12, A-6.2-13, A-6.2-14, A-6.2-15, and A-6.2-17) have been sufficiently resolved. Responses to Audit Issues A-6.2-15 and A-6.2-17 were docketed (ML26035A332 and ML26061A096, respectively). The applicant has shown that the calculated containment peak pressure has sufficient margin to the containment design pressure for all limiting DBAs analyzed. The applicant also has shown that containment pressure is reduced to less than 50 percent of the peak calculated pressure for all DBAs analyzed within 24 hours of the postulated event initiation. The staff concludes that the BWRX-300 containment design referenced in the PSAR conforms to applicable regulatory requirements, and the technical issues deferred to subsequent resolution for the final design for the FSAR at the OL stage do not impact the acceptability of the PSAR for the CP approval. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.2.1.2 Containment Sub-Compartments

6.2.1.2.1 Introduction

The topical reports NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," (Reference 6.2-2) and NEDC-33911P-A "BWRX-300 Containment Performance," (Reference 6.2-3) identify the methodology and regulations to address the sub-compartment differential pressures arising from a DBA. PSAR 6.2.1.2.1 states that large high-energy piping is not routed through containment sub-compartments below the RPV and the containment head area above the refueling bellows seal. PSAR Sections 1.2.2 and 6.2.5 note that BWRX-300 design employs a dry, nitrogen-inerted containment that has limited number of sub-compartments with large openings.

6.2.1.2.2 Regulatory Evaluation

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 6.2.1.2, "Sub-Compartment Analysis," Revision 3, and are summarized below:

- (1) GDC 4, as it relates to the design of containment internal compartments to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. The containment internal compartments shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.
- (2) GDC 50, as it relates to the design of the containment internal compartments to ensure that the reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its

internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

6.2.1.2.3 Technical Evaluation

Specific SRP acceptance criteria to meet the relevant requirements of the NRC regulations identified above for the "Sub-Compartment Analysis," are evaluated as follows for the review described in this SER section.

PSAR Section 6.2.1.2.1 provides information on containment sub-compartments design, and states that the sub-compartment boundaries are not subject to large transient loads resulting from postulated pipe breaks inside the sub-compartments, and the pressure differential across sub-compartment boundaries for the most limiting case is found to be small. It also states that the design of the sub-compartment includes the differential pressures resulting from pipe breaks outside the sub-compartments, and the resulting loads on the sub-compartment boundaries are not large in magnitude. However, the PSAR does not include any sub-compartment analysis results for the referenced containment design and rather refers to NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," (Reference 6.2-2) that demonstrates the use of GOTHIC code and methodology for BWRX-300 sub-compartment analysis.

The staff reviewed and approved the BWRX-300 containment sub-compartments analysis methodology, as a part of NEDC-33922P-A. However, the PSAR conclusion about the pressure differential across the sub-compartment boundaries being small for the most limiting case needs to be justified by a bounding postulated DBA analysis of the pipe breaks inside and outside the sub-compartments of the referenced design. For this purpose, the staff needed to evaluate the actual PSAR sub-compartment design results under the postulated DBAs, to validate the PSAR statements and make the GDC 50 and GDC 4 safety findings by verifying the design details against the following SRP Section 6.2.1.2 acceptance criteria about the sub-compartment nodalization, initial atmospheric conditions, vent critical flow, and the differential pressure design margin.

- Per SRP Acceptance Criterion#1, sub-compartment nodalization schemes should be chosen so that there is no substantial pressure gradient within a node. Following the guidelines of NUREG-0609, a nodalization sensitivity study needs to be performed to verify the nodalization scheme by increasing the number of nodes until the peak calculated pressures converge to small resultant changes. Table 1.9-6 documents the conformance of the acceptance criterion and cites PSAR Section 6.2.1.2, but no related information is presented in the PSAR.
- Per SRP Acceptance Criterion#2, the initial atmospheric conditions within a sub-compartment should maximize the resultant differential pressure. An acceptable model would assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. If the assumed initial atmospheric conditions differ from this model, the selected values should be justified by the applicant. Table 1.9-6 documents the conformance of the acceptance criterion and cites PSAR Section 6.2.1.2, but no related information is presented in the PSAR.
- Per SRP Acceptance Criterion#3, assumptions with regard to the distribution of M&E release should be biased towards maximizing the sub-compartment pressure. The vent flow behavior through all flow paths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100-percent water

entrainment. In addition, the selected vent critical flow correlation should be conservative with respect to available experimental data. Currently acceptable vent critical flow correlations are the "frictionless Moody" with a multiplier of 0.6 for water-steam mixtures and the thermal homogeneous equilibrium model for air-steam-water mixtures. If vent flow paths are used that are not immediately available at the time of pipe rupture, alternate criteria would apply. Table 1.9-6 documents the conformance of the acceptance criterion and cites PSAR Section 6.2.1.2, but no related information is presented in the PSAR.

- Per SRP Acceptance Criterion#4, for the review of a CP PSAR, a factor of 1.4 should be applied to the peak differential pressure, which is calculated in a manner acceptable to the reviewer for the sub-compartment structure, and the enclosed components for use in the design of the structure and the component supports. Table 1.9-6 documents the conformance of the acceptance criterion and cites PSAR Section 6.2.1.2, but no related information is presented in the PSAR.

A staff audit of the documents furnished by the applicant did not provide the missing information about the BWRX-300 sub-compartment design. The staff issued Audit Question A-6.2-16 to request the applicant to provide the information on how the four sub-compartment acceptance criteria were met for the BWRX-300 design referenced in the PSAR or justify the provision of the required information at the OL stage for the FSAR final design. In the response, the applicant addressed the SRP Section 6.2.1.2 sub-compartment design by stating that "The analysis that demonstrates that there is not a significant pressure differential acting across the sub-compartment boundaries is contained in Section 6.6.1 of NEDC-33922P-A, which is incorporated by reference." The response also emphasizes that "the results in NEDC-33922P-A are applicable to the containment design in the TVA PSAR regarding the differential pressure acting on the sub-compartment boundaries, and these results demonstrate that the differential pressures are small and do not pose a significant constraint on designing the sub-compartments. The applicant further made an SRP Section 6.2.1.2 related commitment by stating "If the final detailed design impacts the key parameters, the differential pressures acting across sub-compartment boundaries will be recalculated using the same method described in NEDC-33922P-A but using the inputs of the final design. The final stress analyses will show that the acceptance criteria for the appropriate service level are met." Therefore, the staff accepts the applicant's future commitment to perform an updated sub-compartment analysis for the final BWRX-300 design for the FSAR and demonstrate meeting the four sub-compartment acceptance criteria.

In the initial Audit Question A-6.2-16 response, the applicant also submitted mark-ups replacing the conformance status of each of the four acceptance criteria in Table 1.9-6 by "N/A - BWRX-300 containment design does not have large high energy lines routed through sub-compartments," and stated that "changes are proposed to PSAR Table 1.9-6 to indicate that SRP Section 6.2.1.2 is not applicable to the BWRX-300 design." The staff found the proposed Table 1.9-6 changes inconsistent with the FSAR future commitments, which would render the SRP Section 6.2.1.2 acceptance criteria not applicable to the BWRX-300 design. So, in a modified response, the applicant withdrew the related proposed mark-up changes to PSAR Table 1.9-6, as the staff suggested, and removed the statement from the response that "SRP Section 6.2.1.2 is not applicable to the BWRX-300 design." This applicant's corrective actions to make SRP Section 6.2.1.2 applicable to the TVA containment design is acceptable to the staff.

Based on engineering judgement, the staff concludes that there is adequate assurance from the response, FSAR commitment, and NEDC-33922P-A demonstration that the issue of meeting

the SRP Sections 6.2.1.2 acceptance criteria for the BWRX-300 containment sub-compartment designs can be considered resolved for the PSAR for the issuance of the CP purposes.

6.2.1.2.4 Conclusion

The staff performed a review of the TVA BWRX-300 sub-compartment analysis information provided in the PSAR against the relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria specified in SRP Section 6.2.2, "Sub-Compartment Analysis," Revision 3. The staff's review of the TVA containment design compliance with the regulatory requirements led to issuing several audit questions to the applicant. Applicant's responses were received, and all related audit issues (A-6.2-10, A-6.2-16, and A-6.2-17) have been sufficiently resolved. The staff concludes that the BWRX-300 containment design referenced in the PSAR conforms to the regulatory requirements listed in the SRP section, and the technical issues deferred to subsequent resolution for the final design for FSAR at the OL stage do not impact the acceptability of the PSAR for CP approval. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

6.2.1.3.1 Introduction

PSAR Section 3.1.5.1 states that the containment design is based on consideration of a full spectrum of postulated accidents that would result in the release of reactor coolant to the containment. These accidents are evaluated using TRACG as a boundary condition to GOTHIC to calculate containment thermal-hydraulic response. These accidents include liquid, steam, and partial (both steam and liquid) breaks. Evaluation of the containment design is based upon enveloping the results of these analyses while accounting for appropriate margin. The most limiting short-term and long-term pressure and temperature responses are evaluated to verify the integrity of the containment structure. The GOTHIC computer methodology for measuring containment response is provided in NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," Revision 3 (Reference 6.2-2).

PSAR Section 15.2.4.6 describes the bounding event selection for the LOCA scenarios. The scenarios developed in Section 15.2.4.6 for LOCA inside the containment are evaluated in Section 15.5.4.5, to demonstrate that the fuel and containment integrity acceptance criteria are met for at least 72 hours using only passive heat removal systems. Breaks in the pipes attached to the RPV may be as large as the complete rupture of the largest steam or FW pipes, or as small as leaks in smaller pipes attached to the RPV. To identify bounding cases, PSAR Chapter 15 presents DBA results for the large and small breaks in both the liquid and steam pipes inside the containment. The largest postulated pipe breaks are in the MS, FW, and ICS lines. All large pipes have RIVs that close to isolate the breaks rapidly (i.e., within 10 seconds) and prevent significant loss of RPV coolant inventory to keep the core covered throughout the event. The high containment pressure setpoint for break isolation is reached within 1 second for large breaks. As DBA breaks inside containment are postulated to occur downstream of the RIVs, the RPV is isolated, and its inventory loss does not threaten fuel integrity in a LBLOCA, as the ICS has ample capacity to remove the decay heat, depressurize the RPV, and maintain fuel cooling for at least 72 hours. LBLOCAs are assumed to be instantaneous, double-ended guillotine breaks of the isolatable large pipes to bound the thermal hydraulic response of the fuel, RPV, and containment. As described in PSAR

Section 15.2.4.6.1, the isolation valves close rapidly so the effect on ICS availability is not significant for large break M&E releases, and one ICS train is sufficient to remove decay heat and depressurize the RPV. As a bounding assumption, only one ICS train is credited for MS and ICS pipe breaks. This assumption is made so that an ICS steam supply break is bounded by an analyzed MS pipe break. The limiting parameters for LBLOCA events are containment pressure and temperature, with containment pressure reaching its peak value around the time RIVs are fully closed and the break is isolated. The largest steam and liquid pipe breaks are the most limiting for containment response because they have the highest M&E release until the break is isolated. Over the long term, the PCCS reduces containment pressure.

Breaks involving small pipes (i.e., inner diameters less than 19 mm) connected to the RPV do not have automatic RIVs and are considered un-isolated breaks. Small un-isolated breaks may also occur in larger pipes, as the leakage detection system that detects breaks in large pipes connected to the RPV may not detect breaks smaller than the area of a 19 mm diameter circle. If the break is not detected, it is assumed to remain un-isolated for 72 hours because no operator action is credited for this duration. Small breaks conservatively credit two of the three isolation condenser trains even though a break of less than 19 mm equivalent diameter on an isolation condenser does not cause degradation in the isolation condenser heat removal rate. The largest of the un-isolated breaks is most limiting with respect to RPV inventory. Peak containment pressure also is higher as the break size is increased for an un-isolated break.

The largest break sizes are the most limiting for both isolatable LBLOCA category and un-isolatable SBLOCA category. In selecting the scenarios to evaluate for the pipe breaks, all sequences in the fault list are assessed with respect to the M&E release to the containment and the RPV inventory. Additional conservative assumptions described in Section 15.5.4.5 and Section 15.5.9.1 of the PSAR were made for the bounding scenarios to reduce the number of analysis cases. Therefore, the bounding scenarios analyzed for pipe breaks in large and small break categories are scenarios bounding all scenarios in that category, but the liquid and steam pipe breaks are analyzed separately. PSAR Section 15.2.4.6 also outlines common features used in selecting the bounding sequences for pipe breaks (e.g., determining whether the breaks are more limiting with or without LOPP). To bound all break locations, the break location is assumed as close to the RPV as possible, right outside the second or outboard RIV, and the total break flow is the sum of break flows from both ends of the break for a MS pipe break.

6.2.1.3.2 *Regulatory Evaluation*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-coolant Accidents (LOCAs)," and are summarized below:

- GDC 50, which requires, in part, that the reactor containment structure shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.
- Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, which provides requirements to assure that all sources of energy during the LOCA have been considered

6.2.1.3.2 Technical Evaluation

Specific SRP acceptance criteria to meet the relevant requirements of the NRC regulations identified above for the “Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs),” are evaluated as follows for the review described in this SER section.

SRP Section 6.2.1.3 Acceptance Criteria pertain to GDC 50 and Appendix K to 10 CFR Part 50. GDC 50 requires the containment structure and associated heat removal system to be designed with margin to accommodate any LOCA such that the containment design leakage rate is not exceeded. A LOCA potentially causes the greatest pressure surge in the containment and release of fission products when compared to any other accident. Because it is the most severe challenge expected, containment must be designed to definitively withstand this accident. Following GDC 50 ensures that containment integrity is maintained under the most severe accident conditions thus precluding the release of radioactivity to the environment. Table 1.9-6 documents the conformance of GDC 50 for the TVA containment design. PSAR Section 6.2.1.3.1 states that the design of the PCS complies with GDC 50, and the methodology for evaluating the effect of potential accidents, including margin, is provided in Section 15.5. Results of the evaluation are summarized in Section 15.7.

Appendix K to 10 CFR Part 50 provides the required and acceptable features of the ECCS evaluation models as well as the M&E release analysis during LOCA. Section I.A of Appendix K provides a comprehensive list of LOCA heat (energy) sources, and the reactor operating history assumptions associated with those heat sources. Since the M&E release analysis for postulated LOCAs is used to design containment and containment sub-compartments such that they will withstand the worst case LOCA, it is critical that all potential energy sources are considered. Following 10 CFR Part 50, Appendix K, would ensure that containment and containment sub-compartments are designed to accommodate all energy sources for the worst case LOCA, thus precluding the potential release of radioactivity to the environment following such a LOCA. However, Table 1.9-6 documents an “alternate approach” to the conformance of the TVA containment design with Appendix K to 10 CFR Part 50 against Acceptance Criterion#1 in SRP Section 6.2.1.3. Besides, PSAR Section 6.3.3.1 states that “As stated in NEDC-33910P-A, the ECCS evaluation model does not use the alternatives provided in 10 CFR Part 50, Appendix K.” No further information is available in the PSAR about this issue.

The staff performed an audit of the engineering calculations and verified some of the input parameters and assumptions used in the BWRX-300 TRACG LOCA base deck and analyses for the M&E release and RPV temperatures. However, the staff could not identify what the “alternate approach” to 10 CFR Part 50, Appendix K, was, as identified in Table 1.9-6 in conformance of the PSAR containment design with SRP Section 6.2.1.3, “Acceptance Criterion No.1.” To make a safety finding that the containment structure and heat removal system are designed to withstand postulated accident conditions without exceeding the design leakage rate to meet the SRP Section 6.2.1.3 acceptance criteria, the staff needed to evaluate the alternate method with respect to the conservatism in the M&E release calculations for the BWRX-300 containment design, and any safety-significant deviations from Appendix K to 10 CFR Part 50. The staff issued Audit Question A-6.2-17 to request the applicant to clarify how they met the guidance in the acceptance criterion using the “alternate approach,” and provide any supporting information needed in this regard. The applicant responded that the “alternate approach” in Table 1.9-6 means that “the ECCS evaluation model does not use the alternatives provided in 10 CFR 50 Appendix K,” and submitted a Table 1.9-6 change mark-up with the A-6.2-17 response citing PSAR Section 6.3.3.1 to reflect that. However, contrary to the

response, PSAR Section 6.3.3.1 does not provide any information regarding the “alternate approach” used in lieu of Appendix K.

TR NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” Revision 3 (Reference 6.2-2), which was approved by NRC for the analysis method used for the thermal hydraulics performance of the containment design, is incorporated by reference in the PSAR (Reference 1.6-2). Therefore, the PSAR containment safety design needs to follow the LTR methodology for evaluating the containment structure and heat removal systems to withstand the effect of potential accidents. However, while Page 2 of the approved LTR NEDC-33922P-A clearly lists 10 CFR Part 50, Appendix K, as one of the applicable regulations, it is apparently not used for the PSAR Chapter 6 containment safety analyses. Therefore, to make a safety finding for the SRP Section 6.2.1.3, Acceptance Criterion No.1, the staff needed more clarity on what the PSAR has done differently for the containment safety analyses than what was approved in NEDC-33922P-A in regards to the sources of energy and conservatisms used for calculating the LOCA M&E release into BWRX-300 containment. However, considering the more than 20 percent pressure margin demonstrated in the PSAR for the containment design that far exceeds the 10 percent SRP guidance, it is unlikely for any deviations involved in using the “alternate approach” to Appendix K, to erode away such a significant containment pressure margin in the PSAR design. Therefore, a complete resolution of this issue can be deferred to the final design in the FSAR at the OL stage. Therefore, the staff concludes that a CP can be issued, but the applicant would need to demonstrate that requirements for 10 CFR Part 50, Appendix K, would be met for the final design referenced in the FSAR by using the containment evaluation methodology that was approved as a part of NEDC-33922P-A. The applicant also would need to provide assurance that all applicable energy sources and conservative biases approved for the limiting DBAs in the LTR would be considered for the FSAR containment M&E release calculations.

6.2.1.3.3 *Conclusion*

The staff performed a review of the TVA BWRX-300 M&E release analysis information provided in the PSAR against the relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria specified in SRP Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-coolant Accidents (LOCAs),” Revision 3. The staff’s review of the TVA containment design compliance with the regulatory requirements led to issuing several audit questions to the applicant. The applicant’s responses were received, and all related audit issues (A-6.2-10 and A-6.2-17) have been sufficiently resolved. The staff concludes that the BWRX-300 containment design referenced in the PSAR conforms to regulatory requirements listed in the SRP section, and technical issues deferred for subsequent resolution in the final design for the FSAR at the OL stage do not impact the acceptability of the PSAR for the CP approval. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.2.1.4 *Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment*

6.2.1.4.1 *Introduction*

PSAR Section 6.2.1.4, “Mass and Energy Release Analysis for Secondary Steam Rupture (PWR),” states that the BWRX-300 design does not use a secondary system for FW or steam production.

6.2.1.4.2 *Regulatory Evaluation*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in the SRP Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," and are summarized below:

General Design Criteria (GDC) 50, as it relates to providing sufficient conservatism in the mass and energy release analysis for postulated pressurized-water reactor (PWR) secondary system pipe ruptures to ensure the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

6.2.1.4.3 *Technical Evaluation*

The SRP Section 6.2.1.4 acceptance criteria as identified in the above to meet the relevant requirements of the NRC regulations for the "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," are to be followed for the evaluation in the present SER section. However, PSAR Section 6.2.1.4, "Mass and Energy Release Analysis for Secondary Steam Rupture (PWR)," states that the BWRX-300 design does not use a secondary system for FW or steam production. Table 1.9-6 also documents that the Acceptance Criteria#1, 2, and 3 in SRP Section 6.2.1.4 relate to the sources of energy, M&E release rate, and single failure analysis for postulated secondary system pipe ruptures and are not applicable to BWRs. The staff agrees that SRP Section 6.2.1.4 applies the requirements of GDC 50 only to postulated PWR secondary system pipe ruptures to assure that M&E inputs are appropriately conservative for containment integrity. Therefore, the staff concludes that as the SRP Section 6.2.1.4 acceptance criteria are only applicable to the secondary system pipe ruptures in PWRs, they do not apply to BWRX-300. Therefore, the review interfaces of SRP Section 6.2.1.4 with other SRP sections are also not applicable.

6.2.1.4.4 *Conclusion*

The staff concludes that as the BWRX-300 design does not employ a secondary system for FW or steam production, the SRP Section 6.2.1.4 acceptance criteria do not apply to the BWRX-300 design. The staff has no outstanding concerns.

6.2.1.5 *Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System (PWR)*

6.2.1.5.1 *Introduction*

PSAR Section 6.2.1.5, "Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System (PWR)," states that the BWRX-300 design does not rely on containment pressure to achieve emergency core cooling, as discussed in NEDC-33911P-A.

6.2.1.5.2 *Regulatory Evaluation*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," Revision 3.

The acceptance criteria are based on the relevant requirements of 10 CFR 50.46, which allows either an acceptable ECCS evaluation model that realistically describes the behavior of the reactor during LOCAs or an ECCS evaluation model developed in compliance with 10 CFR Part 50, Appendix K. Therefore, the analysis is acceptable if it meets the following requirements, as applicable:

- (1) 10 CFR 50.46(a)(1)(i), with respect to an acceptable ECCS evaluation model that realistically describes the behavior of the reactor during LOCAs
- (2) 10 CFR 50.46(a)(1)(ii), with respect to an ECCS evaluation model developed in compliance with 10 CFR Part 50, Appendix K, paragraph I.D.2, which requires that the containment pressure used for evaluating cooling effectiveness during reflood and spray cooling should not exceed pressure calculated conservatively for that purpose

6.2.1.5.3 *Technical Evaluation*

SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," provides guidance for compliance to 10 CFR 50.46 for the performance of the ECCS to reflood the core following a LOCA and the associated analyses of the minimum containment pressure possible during the time until the core is reflooded. As containment minimum pressure directly affects the ECCS performance, its calculation and analysis are a part of the ECCS performance evaluation.

Per Acceptance Criterion#1, to meet the requirements of 10 CFR 50.46(a)(1)(i), the model to determine minimum containment pressure for ECCS studies should comply with RG 1.157 that provides specific acceptable containment pressure models for ECCS performance analysis. 10 CFR 50.46(a)(1)(i) requires the plants to have an ECCS, with cooling performance evaluated for the most severe postulated LOCA. 10 CFR 50.46(a)(1)(i) allows the ECCS evaluation to use a realistic model for the reactor coolant system (RCS) during a LOCA. However, Table 1.9-6 documents that the containment pressure model's compliance with 10 CFR 50.46(a)(1)(i) is not applicable to BWR designs. PSAR Section 6.2.1.5 states that "the BWRX-300 design does not rely on containment pressure to achieve emergency core cooling, as discussed in NEDC-33911P-A." The staff also noted that, as described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.3-1], the BWRX-300 design employs the RPV isolation valves and the ICS to perform the ECCS design functions. LBLOCAs become isolated before containment pressure could affect the ECCS design functions. The safety analysis assumes that the small pipe breaks do not need to be isolated and are mitigated by the elevated placement of the small nozzles on the RPV assembly compared to the top of active fuel, containment size, and the IC heat exchanger size, which helps depressurize the RPV such that the RPV and containment pressures equalize and the inventory loss is stopped before the core is uncovered. As the coolant inventory available inside the RPV is sufficient and containment and RPV pressures equalize along with the downcomer and the ICS operation, adequate core cooling is maintained for both LBLOCA and SBLOCA events without safety related high pressure injection. Therefore, the staff agrees that the SRP Section 6.2.1.5 acceptance criteria are not applicable to the BWRX-300 design as it does not rely on containment pressure to achieve emergency core cooling.

6.2.1.5.4 Conclusion

Based on the information reviewed and audited, the staff concludes that the SRP Section 6.2.1.5 acceptance criteria are not applicable to the BWRX-300 design that does not rely on containment pressure to achieve emergency core cooling.

6.2.2 Containment Heat Removal Systems

6.2.2.1 Introduction

The BWRX-300 uses a PCCS designed to provide long-term containment heat removal after a DBA. The PCCS is an ESF system that passively transfers heat by natural circulation from the containment atmosphere to the reactor cavity pool and equipment pool that form the ultimate heat sink (UHS) for the PCCS in response to a DBA, as described in Section 9.2.5. PSAR Section 6.2.2.3 discusses applicable regulatory requirements and guidance for the PCCS.

PSAR Figure 6.2-1 illustrates the configuration of the PCCS that consists of three independent trains each with a passive containment cooling pipe array that has piping connections to the equipment pool. The PCCS can achieve its function with the failure of a single train. The major components of the PCCS are the passive containment cooling pipe arrays, supply lines, and return lines. The PCCS is not required to maintain acceptable normal operating containment temperature conditions that are maintained by the containment cooling system. Some of the containment heat is also passively transferred to the reactor cavity pool through the containment dome. The PCCS configuration is the same for normal and off-normal operations, as it requires no actuation to function. When the plant is under normal operation, the equipment pool gate is removed to connect the equipment pool and reactor cavity pool to provide a heat sink for the PCCS to transfer containment heat after a DBA. If normal containment cooling by the containment cooling system becomes inoperable, containment pressure and temperature are maintained within the design limits by shutting down the reactor, depressurizing the RPV, and transferring the containment heat to the equipment pool and reactor cavity pool by the PCCS and containment head. The PCCS is always in service unless portions are manually isolated; that is, there are no active components, electric power, or actuation signals required to initiate or maintain PCCS function.

For the most limiting break inside the containment, the PCCS heat removal capacity is expected to be sufficient to maintain the containment within design conditions for a period of at least 72 hours following the initiating event without reliance on operator action. The PCCS heat removal becomes significant when steam is discharged into the containment following a pipe break. The steam discharged into the containment increases the containment temperature and also the steam content and associated condensation on the outside of the PCCS piping. Heat transferred to the PCCS from the containment is removed by the natural circulation of water in single-phase flow inside the PCCS tubes and is rejected to the subcooled water in the equipment pool. The natural water recirculation is thermally driven by density gradients inside the PCCS tubes with an elevation difference between the supply and return line connections to the equipment pool.

During a postulated accident, the ICS has sufficient capacity to remove most of the decay heat energy during the accident directly from the RPV thereby reducing the heat added to the containment, and the PCCS removes only the energy released from the break into the containment. The energy released to containment from a large pipe break is limited because such breaks are quickly isolated by RIV closures described in Section 5.4.3. Energy released

to containment from un-isolated small breaks is limited due to the small break size. In addition, the ICS depressurizes the RPV making the break flow rate and temperature relatively small within a few hours, which reduces the PCCS heat load. The PCCS removes energy released from the break in containment to maintain pressure and temperature inside the containment within design limits.

6.2.2.2 *Regulatory Evaluation*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 6.2.2, "Containment Heat Removal Systems," Revision 5, and are summarized below:

1. GDC 38, which requires the following:
 - A. The containment heat removal system (CHRS) is capable of rapidly reducing the containment pressure and temperature following a LOCA and maintaining these parameters at acceptably low levels.
 - B. The CHRS performs in a manner consistent with the function of other systems.
 - C. The safety-grade design of the CHRS provides suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished in the event of a single failure.
2. GDC 39, "Inspection of Containment Heat Removal System," as it relates to the design of the containment heat removal system to permit periodic inspection of important components
3. GDC 40, "Testing of Containment Heat Removal System," as it relates to the design of the containment heat removal system to allow periodic testing to ensure system integrity and the operability of the system and active components.
4. 10 CFR 50.46(b)(5), as it relates to the requirements for long-term cooling, including adequate net positive suction head margin in the presence of LOCA-generated and latent debris.

The guidance in SRP Section 6.2.2, "Containment Heat Removal Systems," Revision 5 lists the acceptance criteria to meet the above regulatory requirements.

6.2.2.3 *Technical Evaluation*

Specific SRP acceptance criteria to meet the relevant requirements of the NRC regulations identified above for the "Containment Heat Removal System," are evaluated as follows for the review described in this SER section.

SRP Acceptance Criterion#1 requires that the containment heat removal systems should meet the redundancy and power source requirements for an engineered safety feature. As described in PSAR Section 6.2.2, the PCCS can achieve its function with a single failure and does not require AC or DC power to function and can achieve its function with the failure of a single PCCS train. A staff audit of BWRX-300 GOTHIC containment analysis details has verified that

two PCCS trains were credited to the safety analyses to generate the PSAR containment analysis results presented in PSAR Chapter 15. The evaluation of the PCS response to postulated accidents is provided in PSAR Section 15.5. The staff noted that Figure 15.5-104 shows that the calculated containment pressure is rapidly reduced to less than 50 percent of its peak value within a couple of hours for the limiting conservative case of the DBA of the large MS pipe break. The staff needed to conclude that the PCCS can achieve its safety function with a single failure of one PCCS train, and the containment heat removal system redundancy and power source consistent with ESF design conform to the requirements of GDC 38. As described in Section 6.2.1, several audit questions the staff issued (A-6.2-10, A-6.2-11, A-6.2-12, A-6.2-14, and A-6.2-17) in this regard have been resolved. The staff concludes that the information provided in the PSAR and the resolution of staff concerns through the audit show that the containment design referenced in the PSAR meets the system safety function for the limiting large break and small break containment pressures being maintained at acceptably low levels after the DBA.

The BWRX-300 design does not rely on any ESF system in addition to PCCS to remove heat from the containment under the DBA conditions. Therefore, it is acceptable to the staff that no containment spray system or containment fan cooler system are used in the BWRX-300 design to accomplish the GDC 38 safety function requirement; therefore, no supporting analysis is needed, per RG 1.82, to demonstrate that the design of spray system avoids pump damage due to cavitation caused by insufficient net positive suction head margin. Likewise, no demonstration of containment fan cooler performance capability is required. Therefore, the staff agrees with the related contents of PSAR Table 1.9-6 and concludes that SRP Acceptance Criteria#2 and #3 about the containment spray system, and SRP Acceptance Criteria#4 about the fan cooler system do not apply to the BWRX-300 containment design.

Meeting SRP Acceptance Criterion#5 of SRP Section 6.2.2 involves evaluating the potential for heat removal surface fouling over the life of the plant and its impact on the containment heat removal system capacity of the heat exchangers to satisfy GDC 38. PSAR Table 1.9-6 suggests that PSAR Sections 6.2.2 and 9.4.7 provide information on how Acceptance Criterion#5 is met. However, the staff was not able to locate any fouling discussion or analysis results in the PSAR. The PSAR provides no information regarding the fouling development on the surfaces of the ICS and PCCS and their impacts on BWRX-300 containment thermal hydraulic response. Additionally, initial staff audit of the engineering calculations and Excel sheets in response to Audit Question 6.2-9 did not provide any information in this regard. The staff needed to determine whether any provisions are included in the referenced design to prevent or address surface fouling or whether surface fouling has been considered in establishing the heat removal capability of the heat exchangers. Therefore, the staff issued Audit Question A-6.2-12 to request the applicant to provide information on fouling evaluation for GDC 38. Based on the applicant's response and auditing a document on the PCCS performance characteristics sensitivity study, the staff ascertained the following information.

- Thicknesses and thermophysical properties of the crud and paint layer, as well as adequate details of the corresponding GOTHIC model inputs to account for the insulating effect of the crud deposition on the PCCS tube inside and paint outside, were provided. The staff noticed that accounting for crud on the PCCS inside surface has an insignificant impact on PCCS performance as well as the containment response. The staff also accepts that the outer surface of the PCCS does not have paint, and crud is not considered on the PCCS outer surface as it is not submerged in water.
- The applicant's response also addressed fouling on the surfaces of the isolation condensers through the conservatism of the assumed ICS heat removal rate at the rated pressure. The

response summarized the fouling sensitivity analysis results showing the reduction of ICS heat removal rate margin from 35 percent to 18 percent, compared to the PANTHERS-IC test data. The applicant also stated that even using a conservative fouling factor, tube plugging, and larger ICS tube wall thickness, would still have 12 percent available margin to the measured ICS heat removal rate assumed for deterministic safety analyses presented in the PSAR.

The staff concludes that TVA has addressed the staff concerns about the potential adverse impact of fouling development inside the PCCS and ICS tubing on the containment thermal-hydraulic response for the GDC 38 related requirement to meet Acceptance Criterion#5 of SRP Section 6.2.2.

SRP Acceptance Criterion#6 requires that the containment suppression pools in BWRs should be designed to provide a reliable, long-term water source for the ECCS and containment spray system pumps, to satisfy the requirements of GDC 38 and 10 CFR 50.46(b)(5) regarding the long-term spray system and ECCS water recirculation. PSAR Section 6.2.2.3 states that the PCCS is designed to passively provide long-term containment heat removal after a DBA, and the removal of containment heat in the BWRX-300 design does not require a containment spray system, sump, external water source for ECCS function, or active components. Therefore, the staff concludes that design consideration of (1) suppression pool hydraulic performance, (2) evaluation of potential debris generation and associated effects including debris screen blockage, (3) residual heat removal and containment spray system pump performance under postulated post-LOCA conditions, and (4) impacts of debris penetrating strainers on long-term cooling capability of the core is not required for the BWRX-300 design. The staff agrees that the guidance provided in RG 1.82 concerning water sources for long-term recirculation cooling following a DBA is not applicable to the BWRX-300 design. Therefore, the staff concludes that Acceptance Criterion#6 for the long-term recirculation of water through the ECCS and containment sprays to ensure cooling capability does not apply to the BWRX-300 design, as is documented by Table 1.9-6.

Per SRP Acceptance Criterion#7, in meeting the requirements of GDCs 39 and 40 regarding the inspection and testing of the containment heat removal systems, the design should provide for periodic inspection and operability testing of the systems and system components such as pumps, valves, duct pressure-relieving devices, and spray nozzles. PSAR Section 6.2.2.3 provides the evaluation of compliance with GDC 39 and GDC 40 as pertains to periodic inspection and testing of the PCCS. The staff concludes that the GDC 39 periodic inspection requirements would only be applicable to the PCCS, as no torus, sumps, and spray nozzles are used in the BWRX-300. The PSAR states that the PCCS is designed to permit periodic inspection of important components to assure system integrity and capability, and the PCS has provisions for personnel access and habitability during plant outages to perform maintenance, inspections, and tests. PSAR Section 3.1.4.10 has provided a detailed evaluation of GDC 39 for inspection of the BWRX-300 containment heat removal system. Compliance with GDC 40 requires that containment heat removal systems design should permit periodic pressure and functional testing to ensure leak-tight integrity, performance of active components, and overall system operability, thereby protecting the safety function of the containment as an engineered safety feature. The PSAR also states that the PCCS does not require active components or electric power to function, and the system is designed such that each train can be isolated and pressure tested during an outage. PSAR Section 3.1.4.11 has provided a detailed evaluation of GDC 40 for the BWRX-300 inspection of containment heat removal system. Table 3.1-1 provides compliance information for GDC 39 and GDC 40. The staff concludes that sufficient

information has been included in the PSAR to satisfy Acceptance Criterion#7 for protecting the safety function of the containment as an ESF for CP approval purposes.

Per SRP Acceptance Criterion#8, instrumentation should be provided to monitor the performance of the containment heat removal system and its components under normal and accident conditions, to satisfy the system design requirements of GDC 38. The instrumentation should determine whether the system is performing its intended function or whether a system train or component is malfunctioning and should be isolated. PSAR Section 6.2.2.5 states that no actuation signals are required for PCCS, whose availability is indicated by equipment pool level and temperature associated with the Fuel Pool Cooling and Cleanup System (FPC), which is described in Subsection 9.1.3. Valving is provided for PCCS system isolation with MCR indication. PSAR Section 7.3 addresses instrumentation and control functions. The staff concludes that the sufficient information has been included in the PSAR to satisfy Acceptance Criterion#8 for PCCS instrumentation, for the CP approval purposes.

6.2.2.4 Conclusion

The staff performed a review of the applicant's PCCS information provided in the PSAR against the relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria specified in SRP Section 6.2.2, "Containment Heat Removal Systems," Revision 5. The staff's review of the applicant's containment design compliance with the regulatory requirements led to issuing several audit questions to the applicant. The applicant's responses were received, and all related audit issues (A-6.2-15 (ML26035A332) and 6.2-17 (ML26061A096)) have been sufficiently resolved. The staff concludes that the BWRX-300 containment design referenced in the PSAR conforms to the regulatory requirements listed in the SRP section, and the technical issues deferred to subsequent resolution for the final design for FSAR at the OL stage do not impact the acceptability of the PSAR for CP approval. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.2.2.5 References

[6.2-2] LTR NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022.

[6.2-3] NEDC-33911P-A "BWRX 300 Containment Performance," GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, January 2022.

[6.2] NEDC-33910P, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection"

6.2.3 Secondary Containment Functional Design

The BWRX-300 does not employ secondary or dual containment. Therefore, this section is not applicable for CP review.

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6.2.4 Containment Isolation Function

6.2.4.1.1 Introduction

This section identifies and describes the containment isolation design that will be used to maintain an essentially leak-tight barrier to radionuclide release from postulated accidents while allowing the passage of fluids necessary to mitigate such events. PSAR Section 6.2.4, "Containment Isolation Function," describes the CIVs, which are the major components that support the containment isolation function, including CIVs associated with (1) lines connecting to the RCPB, (2) lines connecting to the containment atmosphere, and (3) closed systems.

Mechanical systems with lines penetrating the containment boundary support containment integrity by being capable of isolating and limiting the radionuclide release from a postulated accident. These systems perform either a Safety Category 1 (SC1) or Safety Category 2 (SC2) containment isolation function. To support isolation of fluid lines penetrating containment, CIVs are included in each process system line and instrumentation line.

PSAR Tables 6.2-2 and 6.2-3 provide the CIV configuration for containment isolation of process lines and instrument lines, respectively. PSAR Table 6.2-4 provides the list of the CIVs along with its open or closed position for normal and off-normal conditions. CIVs are designed with valve actuators having positive mechanical means (i.e., without power, motive force, or operator actions) to ensure that upon automatic actuation or a loss of signal or control power, the valves are maintained in the required post-accident valve position of greater safety.

Containment isolation is primarily achieved through automatic or remote manual actuation, except for passive devices like blank flanges, rupture disks, and locked closed valves. Automatic CIVs are designed so they cannot reopen simply by resetting the isolation signal; a deliberate operator action is required after the signal is cleared. Most CIVs close upon receiving isolation signals from the SC1 and SC2 instrumentation and control systems and are designed to fail closed if air or control signals are lost. The exception is the CIS over-pressure vent line isolation valve, which fails open to allow containment venting during a design extension condition (DEC) over-pressurization event.

6.2.4.1.2 Regulatory Evaluation

In accordance with 10 CFR 50.34, "Contents of Applications; Technical Information," GDC 54, GDC 55, GDC 56 and GDC 57, piping systems penetrating the primary containment must be provided with isolation capabilities that reflect their importance to safety of isolating these piping systems.

The following NRC regulations and guidance documents contain the relevant requirements for this review and associated guidance:

- In GDC 1, the NRC requires, in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- In GDC 2, the NRC requires, in part, safety-related SSCs to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and these SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids.

- In GDC 4, the NRC requires, in part, that safety-related SSCs shall be appropriately protected against dynamic effects and designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal operation, maintenance, testing, and postulated accidents.
- In GDC 16, the NRC requires, in part, reactor containment and its systems establish an essentially leak-tight barrier against the uncontrolled release of radioactive materials to the environment.
- In GDC 54, the NRC requires, in part, that piping systems penetrating the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect their importance to safety and as it relates to designing such piping systems, a capability to periodically test the operability of the isolation valves and associated apparatus and determine if valve leakage is within acceptable limits.
- In GDC 55 and GDC 56, the NRC requires, in part, that lines penetrating the primary containment boundary as parts of the RCPB (GDC 55) or as direct connections to the containment atmosphere (GDC 56) shall be provided with isolation valves as follows:
 - one locked-closed isolation valve inside and one outside containment
 - one automatic isolation valve inside and one locked-closed isolation valve outside containment
 - one locked-closed isolation valve inside and one automatic isolation valve outside containment
 - one automatic isolation valve inside and one outside containment
- In GDC 57, the NRC requires, in part, that lines that penetrate the primary containment boundary and are neither part of the RCPB nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment and located as close to the containment as practical.
- 10 CFR 50.34(f)(2)(xiv), includes additional requirements for containment isolation systems.
- 10 CFR 50.34(f)(2)(xv), includes additional requirements on containment purging/venting capabilities.
- In 10 CFR 50.63(a)(2), the NRC requires, in part, that appropriate containment integrity is maintained in the event of an SBO for a specified duration.
- SRP Section 6.2.4, “Containment Isolation System,” provides the criteria to meet the relevant regulatory requirements as well as interfaces with other sections.
- BTP 6-4, “Containment Purging During Normal Plant Operations,” provides the NRC staff’s technical position pertaining to system lines which can provide open paths from the containment to the environs during normal plant operation.
- Regulatory (RG) 1.141, *Containment Isolation Provisions for Fluid Systems*, describes acceptable methods for complying with the Commission’s requirements for isolation of fluid systems that penetrate containment.
- RG 1.11, *Instrument Lines Penetrating the Primary Containment Reactor Containment*, defines an acceptable basis to implement the intent of GDCs 55 and 56, regarding instrument lines.

- RG 1.155, *Station Blackout*, describes acceptable methods for complying with the Commission's regulation that requires nuclear power plants to be capable of coping with a station blackout for a specified duration.

6.2.4.1.3 Technical Evaluation

The NRC staff reviewed the CRN-1 preliminary design provisions that will permit containment isolation in accordance with the guidance of SRP Section 6.2.4 and the requirements listed above. The applicant states that the containment isolation function is an ESF that supports the PCS as an essentially leak-tight barrier to an uncontrolled radionuclide release. As such, the containment isolation function, including provisions for control, indication, and performance under loss/restoration of power conditions, is instrumental in maintaining containment barrier integrity and assuring essential flow paths for reactor core cooling. The applicant further states that the design conforms to the listed SRP sections and meets the listed regulatory requirements. The results of the staff's review are provided below.

In PSAR Section 3.9, the applicant described that CIVs are designed to ASME Code, Section III, and PSAR Table 3A-1, "Preliminary BWRX-300 Component Classification List," indicate that CIVs meet Quality Group A or B specifications, which is consistent with RG 1.26, *Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants*. In addition, as indicated in PSAR Table 3A-1, CIVs are designed to meet seismic Category 1 requirements, which is consistent with the guidance in RG 1.29, *Seismic Design Classification*. Furthermore, the NRC staff notes that the CIV area is also protected from natural phenomena hazards (e.g., earthquakes, winds, tornadoes, and floods) by the reactor building (a seismic Category 1 structure). Therefore, the NRC staff finds that the CRN-1 design is consistent with the requirements of GDCs 1 and 2, as it relates to containment isolation.

In PSAR Section 6.2.4 the applicant indicated that the CIVs and barriers are designed to accommodate the effects of and are compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (including LOCAs). In addition, PSAR Section 3.1 states that for piping in containment areas, the probability of fluid system piping rupture is demonstrated to be extremely low under conditions consistent with the design basis for the piping, such that line breaks do not need to be postulated for the purpose of the evaluation of potential dynamic effects. Accordingly, SC1 SSCs do not require protection against dynamic effects, including the effects of pipe whipping, jet impingement, and blast wave and blowdown loads that may result from piping failure.

The applicant described missile protection in PSAR Section 3.5, "Missile Protection"; protection against dynamic effects in PSAR Section 3.6, "Protection against Dynamic Effects Associated with Postulated Rupture of Piping," and environmental conditions in PSAR Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." The staff confirmed that the safety-related containment isolation function was appropriately identified in these PSAR sections. The staff's finding on GDC 4 is primarily made in these interfacing review sections (e.g., 3.5, 3.6, and 3.11 of this SER). Based on the review discussions above, the NRC staff finds that the containment isolation design is consistent with GDC 4, because environmental conditions and dynamic effects have been appropriately considered in the design of the safety-related containment isolation function.

An essentially leak-tight SCCV encloses the RPV, including the majority of the RCPB and other branch connections for the Nuclear Boiler System and includes containment penetrations

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with essentially leak-tight isolation design features including CIVs, blind flanges, hatches, and electrical penetrations. These containment isolation components are designed consistently with staff guidance discussed above (i.e., GDC 1, 2, and 4) in this section and function to prevent the release of radioactive materials to the environment. The containment isolation components also are designed to support containment leakage testing in order to verify the leak-tight integrity of the containment isolation function. Furthermore, the design, qualification, and testing of the isolation valves is addressed in PSAR Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." The NRC staff's review of leakage testing, and valve design are found in Sections 6.2.6 and 3.9.6, respectively, of this safety evaluation report. Based on the review discussions above, the NRC staff finds that the containment isolation design is consistent with GDC 16 to provide an essentially leak-tight barrier against the uncontrolled release of radioactive materials to the environment.

Upon receipt of a containment isolation signal, CIVs move to their post-accident configuration as indicated in PSAR Table 6.2-4. The isolation configuration is redundant where appropriate, consistent with RG 1.141. In PSAR Chapter 7, the applicant discussed the parameters sensed for the initiation of a containment isolation signal (see PSAR Table 7.3-1, "Safety Category 1 Instrumentation and Control Functions and Initiation Signals"). The staff finds that the diversity of parameters that are sensed for isolating containment is sufficient to ensure timely isolation of all nonessential piping penetrations, and consistent with the requirement for reliable isolation capability.

As described in PSAR Section 6.2.4, CIVs have individual leakage testing performed to support the overall requirement to verify containment leakage is within the allowable limits. Process system lines penetrating containment contain test connections to support leakage testing of the CIVs. Closure of CIVs for leakage test is accomplished by normal means and without preliminary exercising or adjustment. Additional discussion on valve leakage is provided in Section 6.2.6 of this SER.

In PSAR Section 5.2.5.2, "Leakage Detection Instrumentation and Monitoring," the applicant described how leakage is monitored using pressure, temperature, level, and radioactivity instrumentation. This supports the conclusion that there are adequate leakage detection provisions to enable the operators to detect leakage and identify lines that should be isolated. Therefore, the staff finds that the provisions for detecting leakage from the lines outside containment are consistent with SRP 6.2.4 Acceptance Criterion 8 for leak detection.

In PSAR Section 6.2.4, the applicant states that valve closure times are established to minimize radiological releases associated with a postulated accident. The applicant further states that minimizing potential effects of containment flooding due to a pipe break is also considered when determining closure times. The NRC staff finds that this is consistent with the guidance contained in SRP 6.2.4, Acceptance Criterion 14 (i.e., rapid isolation of containment).

Discussions below related to TMI requirements also address reliable isolation capability and the safety importance of isolating these piping systems. Based on the discussion above, the NRC staff finds that the containment isolation design is consistent with GDC 54, as it relates to piping systems that penetrate the primary reactor containment to have leak detection, isolation, and containment capabilities with redundancy, reliability, and performance capabilities that reflect the safety importance of isolating these piping systems.

The Nuclear Boiler System, Reactor Water Cleanup System, and Condensate and Feedwater Heating System have inboard integral RIVs and outboard CIVs. The supply lines for Shutdown

Cooling System (SDC) Trains A and B are part of the ICS and are equipped with an outboard CIV and the inboard ICS RIVs. The injection line for the Boron Injection System has an outboard CIV and an inboard CIV. Each of these process lines that penetrate the containment boundary and is part of the RCPB is equipped with two CIVs arranged in series. These valves are independently capable of isolating the line and are designed to automatically isolate with single-failure-proof reliability. For systems using integral RIVs, either RIV can serve as the inboard CIV. CIVs located outside the containment are positioned as close to the containment boundary as practicable. The integral RIV design incorporates redundant components and features to ensure isolation capability. In the event of a worst-case single failure, only one valve would be affected, and the remaining valve within the RIV assembly can still fulfill the isolation function. Automatic isolation valves are designed to assume the safest position upon loss of actuating power. Instrument lines are provided with excess flow check valves located outside containment and are consistent with RG 1.11, and power-operated CIVs have position indication in the MCR, consistent with RG 1.141.

The ICS contains integral RIVs inside containment and a closed loop design outside containment. The NRC staff finds this arrangement consistent with SRP 6.2.4 Acceptance Criterion 5 and is acceptable as an "other defined basis" because the integral RIVs function as inboard CIVs for both steam supply and condensate return lines and the closed-loop piping outside containment can prevent radioactive releases outside containment. The Control Rod Drive uses internal ball check valves that serve as the automatic isolation CIV inside containment, and a manual isolation valve is provided outside containment. The NRC staff finds this arrangement consistent with SRP 6.2.4 Acceptance Criterion 2 and is acceptable as an "other defined basis" because keeping flow path open is necessary to perform the SCRAM function and comparing risk between a degraded scram function and the negative radioactive fission product releases without an outboard automatic CIV, the SCRAM function demonstrates higher risk significance; therefore, automatic isolation is not required. Based on the discussion above, the NRC staff finds that the containment isolation design is consistent with GDC 55 for each line that is part of the RCPB and penetrates primary reactor containment.

The Containment Monitoring Subsystem (CMon), Chilled Water Equipment, Plant Pneumatics System, Demineralized Water Storage and Distribution Subsystem, and Liquid Waste Management System process lines have inboard and outboard CIVs. Each of these process lines that connect directly to the containment atmosphere and penetrate the containment boundary are equipped with two CIVs arranged in series. These valves are independently capable of isolating the line and are designed to automatically isolate with single-failure-proof reliability. CIVs located outside the containment are positioned as close to the containment boundary as practicable. Automatic isolation valves are designed to assume the safest position upon loss of actuating power. Instrument lines are provided with excess flow check valves located outside containment and are consistent with RG 1.11 and power-operated CIVs have position indication in the MCR, consistent with RG 1.141.

The CIS and the Equipment and Floor Drain System (EFS) have a valve arrangement consisting of two CIVs outside containment because locating a valve inside containment is not practicable. The NRC staff finds this arrangement consistent with SRP 6.2.4 Acceptance Criterion 4 and is acceptable as an "other defined basis" because the CIV nearest to containment and the piping between containment and that CIV are designed to preclude a breach of piping integrity. The CIS overpressure vent line consists of a rupture disk, a rupture disk bypass valve, and a downstream isolation valve. The rupture disk, serving as a sealed-closed barrier, replaces an automatic isolation valve, which is consistent with SRP Acceptance Criterion 6. The downstream isolation valve is normally open, fails open, and does not isolate

automatically. Keeping this valve open ensures the vent path remains available during DEC over-pressurization events, supporting containment integrity and representing the safest valve position. Based on the discussion above, the NRC staff finds that the containment isolation design is consistent with GDC 56 for each line that is connected directly to the containment atmosphere and penetrates the primary reactor containment.

For the BWRX-300 design, The PCCS is the only system that is a closed system inside containment that penetrates the containment boundary and is neither part of the RCPB nor connected directly to the containment atmosphere. The PCCS has three trains that each have one remote manually operated CIV on the supply line and one remote manually operated CIV on the return line. Both CIVs are outside containment and located as close to the containment wall as practical. The NRC staff finds that this valve arrangement is consistent with GDC 57.

6.2.4.1.4 TMI-Related Requirements

The NRC staff notes that in PSAR Section 6.2.4, the applicant commits to following these regulatory provisions and intends to apply them.

The regulation in 10 CFR 50.34(f)(2)(xiv) requires containment isolation systems that do the following (II.E.4.2):

- A. Ensure all non-essential systems are isolated automatically by the containment isolation system
- B. For each non-essential penetration (except instrument lines) have two isolation barriers in series
- C. Do not result in reopening of the CIVs on resetting of the isolation signal
- D. Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation
- E. Include automatic closing on a high radiation signal for all systems that provide a path to the environs

In PSAR Section 6.2.4, the applicant described that CIVs serving in nonessential systems (i.e., not required to prevent, arrest or mitigate the consequences of an accident) are designed to close automatically on a containment isolation signal. Therefore, the requirement in 10 CFR 50.34(f)(2)(xiv)(A) to ensure all nonessential systems are isolated automatically by the containment isolation system is satisfied. Additionally, in PSAR Section 6.2.4, the applicant provided information to conclude that each nonessential penetration has two isolation barriers in series, thereby satisfying the requirements of 10 CFR 50.34(f)(2)(xiv)(B). Furthermore, in PSAR Section 6.2.4, the applicant stated that resetting an isolation signal will not automatically reopen isolation valves until the signal is removed and deliberate operator action is taken (e.g., by operation of a reset switch), which is consistent with RG 1.141. This demonstrates that the 10 CFR 50.34(f)(2)(xiv)(C) requirement is satisfied.

The NRC staff notes that the provisions in 10 CFR 50.34(f)(2)(xiv)(D), requiring containment pressure setpoints as low as compatible with normal operation, and 10 CFR 50.34(f)(2)(xiv)(E), requiring automatic closing on high radiation will both be addressed in the FSAR, and the NRC staff will review compliance as part of that review. Based on the discussion above, the NRC staff finds that the containment isolation design satisfies the 10 CFR 50.34(f)(2)(xiv) items A through C requirements.

The regulation in 10 CFR 50.34(f)(2)(xv) states the following:

Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

Technical Specifications for several operating plants define purge or purging as the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. In addition, staff guidance (BTP 6-4, "Containment Purging during Normal Operations") states that containment purge and vent systems provide plant operational flexibility during normal operations (e.g., facilitate personnel access into containment during reactor power operation). Additionally, whenever containment integrity is required, NUREG-0737 discusses limiting containment purge and venting operation to when there is an established need to improve working conditions to perform a safety-related surveillance or safety-related maintenance procedure.

In PSAR 6.2.4 the applicant indicated that its design does not require or incorporate a purge or venting system function as contemplated by this requirement and, therefore, is not applicable to the BWRX-300 design. Specifically, the PCS is maintained dry and nitrogen-inerted above atmospheric pressure during operation. Based on the Technical Specifications definition of purging, staff guidance contained in BTP 6-4 and the applicant's information presented above (i.e., that the BWRX-300 design does not require or incorporate the capability for containment purging during reactor operation), the NRC staff finds that the 10 CFR 50.34(f)(2)(xv) regulation is not technically relevant to the BWRX-300 design.

6.2.4.1.5 Station Blackout (10 CFR 50.63) and Appropriate Containment Integrity

An SBO means the complete loss of ac electric power to the essential and non-essential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency AC power system). The provisions in 10 CFR 50.63, "Loss of All Alternating Current Power," require, in part, that each plant demonstrate sufficient capacity and capability to ensure that appropriate containment integrity is maintained in the event of an SBO for the specified duration. In accordance with RG 1.155, appropriate containment integrity is ensured by providing the capability, independent of the preferred and blacked out unit's onsite emergency AC power supplies, for valve position indication and closure for CIVs that may be in the open position at the onset of an SBO. As described in PSAR Section 6.2.4, CIVs do not require AC power to function, and actuators are designed to hold their failure position for a minimum of 72 hours without AC power. In addition, position indication in the MCR is provided for power-operated CIVs. This arrangement satisfies the guidance provided in RG 1.155, *Regulatory Position C.3.2.7*, for appropriate containment integrity. The NRC staff finds that the BWRX-300 design satisfies the requirement of 10 CFR 50.63(a)(2) with respect to the ability to maintain appropriate containment integrity because the BWRX-300 design conforms to staff guidance.

6.2.4.1.6 Conclusion

The NRC staff review concludes that the applicant's preliminary containment isolation function design for CRN-1 is consistent with the NRC regulations as set forth in

10 CFR 50.34(f)(2)(xiv), 10 CFR 50.34(f)(2)(xv), 10 CFR 50.63(a)(2) and Appendix A, GDC 1, GDC 2, GDC 4, GDC 16, GDC 54, GDC 55, GDC 56, and GDC 57, for piping systems penetrating the primary containment because the applicant meets the acceptance criteria in SRP 6.2.4 and used acceptable methods in accordance with RG 1.141 for process lines, RG 1.11 for instrument lines, and RG 1.155 for SBO.

Based on the design features described above and its evaluation, the staff concludes that it is acceptable for CP. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.2.5 Combustible Gas Control in Containment

6.2.5.1.1 Introduction

This section identifies and describes the combustible gas control in containment design that will be used to limit scenarios that could produce a hydrogen/oxygen combustion event following a postulated accident. PSAR Section 6.2.5, "Combustible Gas Control in Containment," discusses the combustion potential from radiolytic gases and identifies containment features to withstand the effects of DBAs, which includes an inerted containment and a containment monitoring system.

6.2.5.1.2 Regulatory Evaluation

In accordance with 10 CFR 50.34, "Contents of Applications; Technical Information," 10 CFR 50.44, GDC 41, GDC 42, and GDC 43, the reactor containment must be designed to control the combustible gases in the containment following a beyond DBA involving 100 percent fuel clad-coolant reaction or a postulated accident.

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

- In 10 CFR 50.44, the NRC requires, in part, that the containment shall be designed to accommodate hydrogen generation equivalent to a 100 percent fuel clad-coolant reaction, including the capability for ensuring a mixed atmosphere; providing combustible gas control; ensuring equipment survivability; providing containment monitoring equipment; and demonstrating containment structural integrity.
- In GDC 41, the NRC requires, in part, that systems to control hydrogen and oxygen in the reactor containment shall be provided as necessary to reduce and control the concentration of hydrogen and oxygen in the containment atmosphere following postulated accidents; shall be provided with suitable redundancy in components and features such that the system safety function can be accomplished, assuming a single failure; and shall be provided with suitable leak detection, isolation, and containment capability to ensure that system safety function can be accomplished.
- In GDC 42, the NRC requires, in part, that the containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspections.
- In GDC 43, the NRC requires, in part, that the containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing.

SRP Section 6.2.5, "Combustible Gas Control in Containment," provides criteria to meet the relevant regulatory requirements as well as interfaces with other sections. Also, RG 1.7,

Control of Combustible Gas Concentrations in Containment, describes acceptable methods that may be used to comply with 10 CFR 50.44 for the mitigation of combustible gas generated by a beyond DBA.

6.2.5.1.3 Technical Evaluation

The NRC staff reviewed the CRN-1 preliminary design provisions that will mitigate combustible gas generated in containment in accordance with the guidance of SRP Section 6.2.5 and the requirements of 10 CFR 50.44. The NRC regulations in 10 CFR 50.44(c) are applicable combustible gas control requirements for future water-cooled nuclear power reactor designs. The applicant states that the BWRX-300 design employs a dry, inerted containment with limited sub-compartments that have large openings. Radiolytic gases mixed with steam and liquid are released to containment in the event of a LOCA. The combustion potential from these gases is not a concern for large breaks because the rapid closure of the RIVs terminates break flow and the release of radiolytic gases. Radiolytic gases may build up in the containment following un isolated small breaks over time, although the rate of release is small. Analyses demonstrate the sub-compartments are sufficiently mixed and that gases generated by radiolysis do not create a deflagration concern. Evaluation of BWRX-300 containment structural integrity to demonstrate the survivability of the containment under structural loads generated by an accident in which 100 percent fuel clad-coolant reaction accompanied by hydrogen burning will be done as part of the FSAR review, using the methodology described in NEDC-33926P. The NRC staff reviewed the containment design for consistency with 10 CFR 50.44(c) as discussed below.

Mixed Atmosphere

The BWRX-300 containment has limited sub-compartments that have large openings. The sub-compartments are well mixed due to the open connections between containment and the volume below the RPV and containment and the space between the RPV and the biological shield.

NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," provides a discussion and calculations on the radiolytic gases mixed with the steam and liquid that are released to containment in BWRX-300 DBAs. The combustion potential from these gases is not a concern for large breaks because the rapid closure of the RIVs terminates the break flow and the release of radiolytic gases, however, radiolytic gases may build up in the containment following un-isolated small breaks over time. As part of the review of NEDC-33922P-A, the NRC staff confirmed that there will be sufficient mixing in containment sub-compartments and that the calculated combustible gases concentrations inside the containment following un-isolated small breaks are far below the deflagration limits.

Based on above, the NRC staff finds that gases generated by radiolysis are not likely a concern for deflagration in the containment for the BWRX-300 design and the identified containment sub-compartments have sufficient mixing, consistent with guidance of RG 1.7. Therefore, the CRN-1 design meets the requirements of 10 CFR 50.44(c)(1) for ensuring a mixed atmosphere in the containment following DBAs and beyond DBAs.

Combustible Gas Control

The BWRX-300 design employs a dry, inerted containment. NEDC-33911P-A, "BWRX-300 Containment Performance," describes that the BWRX-300 dry containment is nitrogen inerted, and containment inerting is maintained during operation by a CIS. The CIS can

preclude the development of a combustible atmosphere by maintaining an oxygen deficient atmosphere inside containment. The NRC staff's evaluation of the CIS is in Section 9.3.6 of this SER. The NRC staff finds that the CRN-1 design for the oxygen concentration satisfies the guidance of 10 CFR 50.44, which defines an inerted atmosphere as less than 4 percent volume, and is consistent with the guidance of RG 1.7. Therefore, the design meets the requirements of 10 CFR 50.44(c)(2) for combustible gas control to mitigate the risk associated with combustible gas generation attributed to beyond DBAs.

Equipment Survivability

Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and the containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. As discussed above in Combustible Gas Control, the BWRX-300 design establishes and maintains an inert atmosphere within the containment to prevent hydrogen combustion. Furthermore, the BWRX-300 design does not rely upon combustible gas control to maintain safe shutdown and containment structural integrity following a DBA. Therefore, the NRC staff finds that the CRN-1 design meets the requirements of 10 CFR 50.44(c)(3) by maintaining an inerted atmosphere in containment.

Monitoring

The equipment for monitoring hydrogen and oxygen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen and oxygen in the containment atmosphere following a beyond DBAs for accident management, including emergency planning. The BWRX-300 design uses oxygen and hydrogen analyzers to monitor containment oxygen and hydrogen concentrations. The capability to monitor containment hydrogen and oxygen concentration is part of the CMon. The NRC staff's evaluation of the CMon is in Section 6.8 of this SER. The NRC staff finds that the CRN-1 design, regarding the monitoring of hydrogen and oxygen concentrations in containment, is consistent with the guidance of RG 1.7. Therefore, the design meets the requirements of 10 CFR 50.44(c)(4) for monitoring hydrogen and oxygen concentrations.

Structural Integrity

Steel containments must meet the requirements of the ASME Boiler and Pressure Vessel Code (edition and addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Sub-Article NE-3220, Service Level C Limits, considering pressure and dead load alone.

In PSAR Section 6.2.5, the applicant stated that the evaluation of BWRX-300 containment structural integrity to demonstrate the survivability of the containment under structural loads generated by an accident in which 100 percent fuel clad-coolant reaction accompanied by hydrogen burning will be evaluated in the FSAR using the methodology described in NEDC-33926P, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design." Therefore, based on this commitment, the NRC staff finds that the design is consistent with RG 1.7 and meets the requirements of 10 CFR 50.44(c)(5) to demonstrate containment structural integrity through analysis. The staff will review the actual structural integrity evaluation during future licensing activities for CRN-1, as part of the FSAR review.

GDC 41 relates to containment atmosphere cleanup systems being designed to control the concentration of hydrogen and oxygen in the containment atmosphere following postulated accidents. As discussed above, the BWRX-300 design is a dry containment that is nitrogen inerted and maintained during operation by the CIS. Fission products, hydrogen, oxygen, and other substances released from the reactor are contained within the low-leakage containment, and oxygen monitors are installed for monitoring during and after a DBA. The NRC staff finds that the BWRX-300 design, as described in the PSAR, will provide the plant with the capability to monitor and control the concentration of hydrogen or oxygen in the containment atmosphere following postulated accidents to ensure that containment integrity is maintained. Therefore, the NRC staff finds that the CRN-1 design is consistent with the requirements of GDC 41 for monitoring combustible gases in the containment. However, full compliance with GDC 41 will be based on the NRC staff's FSAR review of the containment and combustible gas control and monitoring system, considering the results for DBAs and severe accidents.

GDC 42 relates to containment atmosphere cleanup systems, specifically in this case the CIS, being designed to permit appropriate periodic inspections. The CIS design includes provisions for periodic inspection, operability testing, and leak rate testing of the important components to the CIS. Specific components subject to the periodic inspection of the CIS is discussed in Section 9.3.6 of this safety evaluation report. The NRC staff finds that the CRN-1 design meets the acceptance criteria in SRP 6.2.5 for appropriate periodic inspections of the CIS; therefore, the design is consistent with GDC 42.

GDC 43 relates to containment atmosphere cleanup systems, specifically in this case the CIS, being designed to permit appropriate periodic pressure and functional testing. The CIS design includes provisions for periodic testing to ensure system integrity and operability of the system and active components. Specific details on the periodic testing of the CIS are discussed in Section 9.3.6 of this SER. The NRC staff finds that the CRN-1 design meets the acceptance criteria in SRP 6.2.5 for appropriate periodic testing of the CIS, therefore the design is consistent with GDC 43.

6.2.5.1.4 Conclusion

The NRC staff review concludes that the applicant's preliminary combustible gas control in containment design for CRN-1 is consistent with the NRC regulations as set forth in 10 CFR 50.44 and Appendix A, GDC 41, GDC 42, and GDC 43, for the control of combustible gas concentrations in containment because the applicant meets the acceptance criteria in SRP 6.2.5 and used acceptable methods in accordance with RG 1.7 for the mitigation of combustible gas generated by a beyond DBA.

Based on the design features described above and its evaluation, the staff concludes that the description of the combustible gas control in containment is acceptable for CP. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.2.6 Containment Leakage Testing

6.2.6.1.1 Introduction

This section identifies and describes the containment leakage rate testing design that will be used to verify the leak-tight integrity of the containment. PSAR Section 6.2.6, "Containment

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Leakage Testing,” describes the leak rate tests that are applicable to CRN-1, which includes integrated leakage rate test (ILRT, or Type A), containment penetration leakage rate test (Type B), and CIV leakage rate test (Type C).

6.2.6.1.2 Regulatory Evaluation

In accordance with 10 CFR 50.34, “Contents of Applications; Technical Information,” GDC 52, GDC 53, and GDC 54, reactor containment and piping systems that penetrate the primary containment must be designed so that periodic testing can be conducted to determine that leakage is within acceptable limits.

The following NRC regulations and guidance documents contain the relevant requirements and associated guidance for this review:

- In GDC 52, the NRC requires, in part, that the reactor containment and exposed equipment shall be designed to accommodate the test conditions for the containment integrated leakage rate test (up to the containment design pressure).
- In GDC 53, the NRC requires, in part, that the reactor containment shall be designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and periodic testing at the containment design pressure of penetrations having resilient seals and expansion bellows.
- In GDC 54, the NRC requires, in part, that the piping systems penetrating primary reactor containment shall be designed with a capability to determine if valve leakage rate is within acceptable limits.
- In 10 CFR Part 50, Appendix J, the NRC specifies the leakage rate test requirements and acceptance criteria for preoperational and periodic testing of the leak-tightness of the reactor containment and penetrations.
- SRP Section 6.2.6, “Containment Leakage Testing,” provides the criteria to meet the relevant regulatory requirements as well as interfaces with other sections.
- Regulatory (RG) 1.163, *Performance-Based Containment Leak-Test Program*, describes an acceptable performance-based leak-test program and leakage rate test methods, procedures, and analyses that may be used to comply with “Option B – Performance-Based Requirements” in Appendix J.

6.2.6.1.3 Technical Evaluation

The NRC staff reviewed the CRN-1 preliminary design provisions that will permit containment leakage rate testing in accordance with the guidance of SRP Section 6.2.6 and the requirements of 10 CFR Part 50, Appendix J. The applicant indicated the methodology used for designing BWRX-300 complies with American National Standards Institute/(American Nuclear Society (ANSI/ANS)-56.8-2020, “Containment System Leakage Testing Requirements.” The NRC endorsed this industry standard in RG 1.163, Revision 1, for acceptable use on technical methods and techniques for performing Type A, B, and C tests.

PSAR Section 6.2.6, describes that the tests assure that:

1. Leakage through the containment and systems and components penetrating containment does not exceed allowable leakage rate values as specified in the Technical Specifications.

2. Periodic surveillance of containment penetrations and isolation valves is performed so proper maintenance and repairs are made during service life of the containment, and systems and components penetrating containment.

As part of the containment leakage rate testing program, the applicant will document test reports, the test results, analyses, and structural conditions of tested SSCs.

GDC 52 relates to the reactor containment and exposed equipment being designed to accommodate the test conditions for the containment ILRT (up to the containment design pressure). The applicant has committed to preoperational and periodic Type A testing in PSAR Section 6.2.6.1, "Integrated Leakage Rate Test (Type A)." The ILRT will be performed after construction of the primary containment system and following satisfactory completion of structural integrity testing, to verify that the containment leakage rate does not exceed the design limit. The Type B and C leakage rate tests will also be completed prior to the ILRT preoperational Type A test and the closure of CIVs for Type A test is accomplished by normal means and without preliminary exercising or adjustment. PSAR Table 6.2-1, "Primary Containment System Key Design Parameters," indicates that the containment design leakage rate is limited to less than 0.35 percent of the weight in the containment free volume per day at rated design pressure. PSAR Table 6.2-1 also indicates the containment design pressure to be 60 pounds per square inch gauge (psig), however, in order to establish a point of reference for future in-service leak tests, the preoperational ILRT will be conducted at a pressure consistent with the pressure selected for the periodic in-service leak tests.

Periodic ILRTs will be conducted to ensure containment leakage remains less than the maximum allowable leakage rate (L_a) and to identify adverse trends since previous ILRTs. The BWRX-300 maximum allowable leakage rate is 0.35 percent per day. These periodic tests will be performed at established intervals, after major repairs, and upon indication of excessive leakage and will follow the same methodology and acceptance criteria as the preoperational ILRT. The NRC staff finds that the CRN-1 design complies with 10 CFR Part 50, Appendix J, and is consistent with the guidance of RG 1.163 for Type A testing. Therefore, the design meets the criteria of GDC 52 for integrated leakage rate testing of the reactor containment.

GDC 53 relates to the reactor containment being designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and leakage rate test at the containment design pressure of penetrations having resilient seals and expansion bellows. TVA has committed to preoperational and periodic Type B testing in PSAR Section 6.2.6.2, "Containment Penetration Leakage Rate Test," on containment penetrations incorporating resilient seals, bellows, gaskets, or sealant compounds; airlocks and airlock door seals; equipment and access hatch seals; and electrical penetration canisters. The NRC staff finds that the CRN-1 design complies with 10 CFR Part 50, Appendix J, and is consistent with the guidance of RG 1.163 for Type B testing. Therefore, the design meets the criteria of GDC 53 for inspection, surveillance, and leakage testing of containment penetrations.

GDC 54 relates to piping systems penetrating primary reactor containment being designed with a capability to determine if the valve leakage rate is within acceptable limits. The applicant has committed to preoperational and periodic Type C testing in PSAR Section 6.2.6.3, "Containment Isolation Valve Leakage Rate Test," on CIVs and RIVs that are the inside containment CIV. Type C tests will be performed by local pressurization and applied in the same direction as when the valve is required to perform its safety function, unless it can be determined that the results from tests for a pressure applied in a different direction provides

equivalent or more conservative results. The specific test plan for valves (including RIVs that are CIVs), will be identified in the FSAR, and reviewed by the NRC staff at that time. Each valve to be tested will be closed by normal operation without preliminary exercising or adjustment. The NRC staff finds that the CRN-1 design complies with 10 CFR Part 50, Appendix J, and is consistent with the guidance of RG 1.163 for Type C testing. Therefore, the design meets the criteria of GDC 54 for operability testing of isolation valves in the containment penetrations.

6.2.6.1.4 Conclusion

The NRC staff's review concludes that the applicant's preliminary containment leakage testing design for CRN-1 is consistent with the NRC regulations as set forth in 10 CFR Part 50, Appendix J and Appendix A, GDC 52, GDC 53, and GDC 54, for the reactor containment and piping systems penetrating the primary containment to be capable to conduct periodic leakage rate testing because the applicant meets the acceptance criteria in SRP 6.2.6 and sufficiently described the types of test in accordance with RG 1.163 for a performance-based containment leakage rate testing program.

Based on the design features described above and its evaluation, the staff concludes that it is acceptable for CP. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The applicant states that the containment is designed to provide an essentially leak-tight barrier to the release of radioactivity. As such, it is designed to assure that it is not subject to brittle fracture under loadings during operating, maintenance, testing and postulated accident conditions. The design of the containment boundary with regard to compliance with GDC 16 and GDC 51 is addressed in Section 3.8 of this SER and NEDC-33926P. The evaluation provided in Section 15.5 demonstrates that PCS temperature and pressure after a postulated accident are maintained within design parameters for at least 72 hours without operator action. Based on this design's features, the staff concludes that it is acceptable for CP.

6.3 Emergency Core Cooling System (Isolation Condenser System)

The BWRX-300 design uses passive systems as the ESFs that do not depend on operator actions or external power, to fulfill the safety functions for at least 72 hours after a DBA. The BWRX-300 Isolation Condenser System (ICS) is an ESF that provides a physical barrier against radiological releases to the environment. Reactor pressure vessel heat removal during transient and accident conditions is provided by the ICS. Parts of the ICS are part of the containment boundary in addition to being part of the reactor pressure boundary. PSAR Sections 6.3.1 and 6.3.2 describe the ICS and the proposed design bases for the ICS. The proposed testing and instrumentation requirements are listed in PSAR sections 6.3.4 and 6.3.5. The following SER Sections review the applicant's PSAR following the applicable acceptance criteria in NUREG-800. PSAR Table 1.9-6 tabulates the conformance with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", for PSAR Chapter 6, while PSAR Table 3.1-1 provides additional information about the compliance with the applicable General Design Criteria/Principal Design Criteria involved.

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6.3.1 Design Bases (Isolation Condenser System)

The ICS is not a traditional ECCS system. The primary safety functions of the ICS are emergency core cooling, over-pressure protection, and shutdown cooling. It is part of the pressure boundary and the containment boundary. The ICS has three trains with condensers sitting in a pool outside containment with a rated heat removal capacity of 33.75 MWt (3.8 percent of reactor power) per train. It has a design pressure of 1,500 psig and a design temperature of 598°F, which is the saturation temperature at the design pressure.

6.3.1.1 Introduction

PSAR section 6.3.1 lists the SC1, SC2, and SC3 functions the BWRX-300 ICS needs to perform including decay heat removal, protecting the integrity of the RCPB, detecting breaks in ICS lines, closing the standby gas purge valves, isolation of shutdown cooling, and prevention of non-condensable gas accumulation in the ICS while it is in standby mode. The SC1 classification is equivalent to what the NRC calls safety related. The ICS is part of the RCPB and is the required ASME Boiler and Pressure Vessel Code (BPVC) over-pressure protection system functioning in conjunction with a reactor SCRAM.

6.3.1.2 Conclusion

The description of the design basis in PSAR section 6.3.1 describes the functions for the different defense lines of the BWRX-300 safety system that the ICS is required to meet. The description in the PSAR of the proposed design basis for the ICS has adequate detail to allow the staff to review the ICS system functions and evaluate the adequacy of the system to perform its safety functions. The review criteria and evaluation are provided in SER section 6.3.3.

6.3.2 System Description (Isolation Condenser System)

6.3.2.1 Introduction

PSAR section 6.3.2 describes the ICS system. The system is composed of the steam inlet piping that is connected to the RPV, the condensers, the condensate return line piping that drains condensed liquid to the chimney region, and the standby gas purge line that connects the inlet piping to a steam line. There are isolation valves on the inlet line, exit line, and the standby gas purge line. The condenser sits in a large pool of water that is outside of containment. A simplified diagram of the system is shown in PSAR Figure 6.3-1, and the attachment of the recombiner to a condenser is shown in PSAR Figure 6.3-2.

The steam supply lines are open during normal operation. There is instrumentation to detect pipe breaks so the lines can be isolated. Water will fill the line on the ICS side up to the elevation where the standby gas purge lines connect since the outside surface of the piping is in a pool of cold water. The standby gas purge lines provide a continuous flow of gas to the steam lines. This venting to the steam lines keep radiolysis gases from building up in the inlet lines.

The condensers are made up of upper drums, vertical condenser tubes, lower drums, and a recombiner system with a flow path from the lower to the upper header. The condensers are filled with water before they start to operate. Steam flows into the condensers while they are operating and condenses into a thin liquid film on the tube walls. Some steam will condense on the drum walls in addition to the tube walls. The energy from condensation is conducted through the walls into the pool water. The pool water will boil when the condensation heat flux is high.

The recombiners prevent the buildup of radiolysis gases in the condenser which can degrade the performance of the condenser. The recombiners cannot prevent the buildup of any non-condensable gas that is ingested from the containment if the RPV pressure drops below the containment pressure.

The condensate return lines drain the water to the chimney region of the RPV. The return lines have two condensate return valves in parallel to prevent a single failure from allowing the system to operate. Since the water is cold it can cause additional condensation of the vapor in the chimney region. The condensate return line has a loop seal to prevent the flow of steam up through the return line to the condensers which can degrade the performance of the condenser. The SDC system takes water from the chimney through the Train A and B return lines. The boron injection system injects a borated solution into the ICS Train C drain line.

There are five modes of operation for the ICS. In the normal Standby mode, the gas purge line valves are open and the trains are waiting for a signal to open the condensate return valves and close the gas purge line valves to put them into operation in the In-Service mode. In the In-Service mode, steam flows from the RPV to the condensers and water returns to the chimney region of the RPV through the drain line. The recombiner operates to prevent the buildup of radiolysis gases. In the Shutdown Cooling Operation mode, the SDC system takes water from the chimney region of the RPV through the return lines and into the SDC system. In the out-of-service mode of operation, the condensers are taken out-of-service and cannot remove heat. The SDC system can still take suction water through the return lines in this mode of operation. The boron injection operation mode allows boron to be injected into the Train C return line, which delivers the borated water to the chimney region of the RPV in the case in which the normal reactivity control system is not able to shut down the reactor.

During normal power operations the ICS is in Standby mode. A single train can be placed in operation during a normal shutdown to do performance testing of the ICS. The ICS is in Standby mode during hot shutdown. During cold shutdown, one train of ICS must be operable to take over the heat removal function if the SDC system function is lost.

During off-normal operation, the ICS provides emergency core cooling and over-pressure protection. The condensate return valves open and the gas purge line valves close when the ICS is placed into operation. The gas purge line valves close for all trains when the first train of ICS is placed into operation. The number of ICS trains in service depends on the type and severity of the event. Each train can have a separate initiation point, or they can all initiate together depending on the type and severity of the event. The emergency core cooling function causes the reactor to reduce the pressure by condensing steam and returning the condensate to the RPV. This reduces the break flow for breaks that cannot be isolated and keeps the core from uncovering in the first 72 hours after the break. The isolation functions for trains A and C are in a separate, independent, and diverse SC1 Control system from the isolation function for train B so that a common cause failure cannot spuriously isolate all three trains. There is also a separate system in each train to detect and isolate a break in the train and let the other trains operate.

6.3.2.2 Conclusion

PSAR section 6.3.2 provides a system level description of the ICS and its functional requirements. The description is supported by figures that illustrate important aspects of the system and the combined material is adequate to allow the staff to review the material and

evaluate the adequacy of the system to perform its safety functions. The review criteria and evaluation are provided in SER section 6.3.3.

6.3.3 Safety Evaluation (Isolation Condenser System)

6.3.3.1 Introduction

PSAR section 6.3.3 describes the applicable regulations that the applicant believes are applicable to the ICS and describes how they are met. The ICS must perform its design basis functions in compliance with the applicable Appendix A of 10 CFR Part 50, the applicable GDC, 10 CFR 50.46, post-TMI action items listed in 10 CFR 50.34(f), and other related requirements.

6.3.3.2 Regulatory Evaluation

The ICS has functions that apply to the ECCS function and has additional functions for over-pressure protection and long-term heat removal. The regulatory requirements applicable to the ICS design are applicable GDCs listed in Appendix A 10 of CFR Part 50, the TMI action items listed in 10 CFR 50.34(f), the ECCS requirements in 10 CFR 50.46, the applicable codes and standards in 10 CFR 50.55a, station blackout requirements in 10 CFR 50.63, and ECCS evaluation model documentation requirements in 10 CFR Part 50, Appendix K.

The regulations that apply to the ECCS function are discussed in SRP section 6.3 and include the following:

- GDC 2 as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function.
- GDC 4 as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer).
- GDC 5 as it relates to SSCs important to safety shall not be shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.
- GDC 17 as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded during anticipated operational occurrences and that the core is cooled during accident conditions.
- GDC 27 as it relates to the system design having the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.
- GDC 30 as it relates to the quality of the RCPB and provides means for detecting and identifying the location of reactor coolant leakage.
- GDC 31 as it relates to fracture prevention of the RCPB.
- GDC 32 as it relates to inspection of the RCPB.
- GDCs 35, 36, and 37 as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.

- 10 CFR 50.46, in regard to the ECCS being designed so that its cooling performance is in accordance with acceptable evaluation models, which identifies and accounts for uncertainties in the analysis method and inputs; alternatively, an ECCS evaluation model may be developed in conformance with Appendix K to 10 CFR Part 50.
- TMI Action Plan item II.K.3.18 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(vii) for applicants subject to 10 CFR 50.34(f), with respect to eliminating the need for manual actuation of the BWR ADS to assure adequate core cooling.
- TMI Action Plan item III.D.1.1 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xxvi) for applicants subject to 10 CFR 50.34(f), with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the ECCS outside of the containment that contain or may contain radioactive material following an accident.
- Conformance to TMI Action Plan item II.F.2 of NUREG-0737, which is equivalent to 10 CFR 50.34(f)(2)(xviii), with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling will be addressed in the FSAR.

Additional related sections that apply to the ICS other than the ECCS function include:

- GDC 34—Residual heat removal. The ICS supports decay heat removal such that specified acceptable fuel design limits are not exceeded during postulated accidents as demonstrated in Section 15.5. See Subsection 5.4.7 with regard to the SDC and Chapter 10 for normal methods of residual heat removal.
- GDC 44—Cooling water. The ICS transfers decay heat from the RPV to the UHS during post-accident conditions. Compliance with GDC 44 is addressed in sub-section 9.2.5.
- GDC 45—Inspection of cooling water systems. Compliance with GDC 45 regarding inspection of UHS components is addressed in sub-section 9.2.5.
- GDC 46—Testing of cooling water systems. Compliance with GDC 46 regarding testing of UHS components is addressed in sub-section 9.2.5.
- The criteria for overpressure protection and compliance with those criteria are described in LTR NEDC-33910P-A, Revision 1.

6.3.3.3 *Technical Evaluation*

PSAR section 6.3.3 says that the ICS ECCS function requires that the system remove decay heat and reduce the system pressure to minimize the inventory loss for the breaks that are not isolated. It must reduce pressure and break flow so that the core will not uncover and heat up for the first 72 hours after the LOCA. Calculations documented in Chapter 15 of the PSAR show that the ICS successfully performs that function for small breaks in the steam and liquid spaces. Therefore, the staff finds that the ICS ECCS function is met.

PSAR section 6.3.3 says that the ICS over-pressure protection function requires that the ICS has enough heat removal capability to keep the RCS below the applicable pressure limits for anticipated operational occurrences, DBAs, and anticipated transient without SCRAM. The calculations that demonstrate this are documented in Chapter 15 of the PSAR. The staff review of the calculations is documented in chapter 15 of the SER. The ICS heat removal in those calculations is adequate to keep the RCS pressure below the applicable pressure limits. Therefore, the staff finds that the ICS over-pressure protection function is met.

PSAR section 6.3.3 says that the ICS long term cooling function requires that the ICS can remove decay heat for 7 days following shutdown of the reactor. The ICS pools are the UHS for this function, and they must be sized to contain enough water to perform that function. The Isolation Condenser Pools Cooling and Cleanup System is described in PSAR Section 9.2.8. One of the functions of the system is to maintain the water level in the pool during normal operations to provide sufficient water for 7 days of passive heat removal during off-normal operations. Therefore, the staff finds that the ICS long term cooling function is met.

The BWRX-300 ECCS system is significantly different than operating BWRs so some of the TMI action items listed in 10 CFR 50.34(f) and some of the SRP Section 6.3 review criteria do not apply. The BWRX-300 ICS in conjunction with the reactor vessel isolation valves on the MS line, the FW lines, and the ICS steam supply and condensate return lines function as the ECCS system. The safety strategy for the ECCS system is to detect and isolate breaks for all large pipes and use the ICS to minimize the loss of RPV inventory in smaller breaks so that the core does not uncover and heat up for times less than 72 hours. The PSAR design takes credit for adding inventory using a pumped system powered by a diesel generator after 72 hours.

The ICS, including the recombiner, acts as a passive thermosyphon component once the drain line valves are open to begin operation to provide emergency core cooling or over-pressure protection. The ICS system meets Limitation and Condition #1 and #2 in the staff SER for NEDC-33922P-A by having the recombiner system on the condenser and the loop seal in the ICS condensate return line.

The heat removal capacity of the condensers has been tested at almost full scale during the development of the Simplified Boiling Water Reactor (SBWR) in the PANTHERS test facility. A description of the PANTHERS test facility and a summary of the results can be found in the TRACG Qualification Reports for SBWR (NEDC-32725P). PANTHERS was a full-scale test facility for the SBWR condensers, which were smaller than the Economic SBWR condensers that the BWRX-300 uses. The condensers were tested over a wide range of pressures from ~0.3 MPa up to ~8.1 MPa using pure steam and a mixture of steam and non-condensable gases. The tests with steam and non-condensable gases show how the non-condensable gas builds up to high concentrations and degrades the condensation on heat removal. The autocatalytic recombiner functionality is new to the BWRX-300 isolation condensers. GEV will conduct a test program to validate the functionality of the autocatalytic recombiner. The test program is described in NEDC-34349P, "Darlington New Nuclear Project (DNNP) Joint Report on GE Hitachi's Containment Evaluation Method: Implementation of Licensing Considerations."

The staff found that the ICS is required to perform at RPV pressures lower and higher than the pressures in the PANTHERS tests during some Chapter 15 DBA and DEC events. The lower pressure conditions are covered by the PANTHERS PCCS testing that used a heat exchanger similar to the ICS heat exchanger but in lower pressure conditions for the SBWR and Economic SBWR containments. The high-pressure conditions (>8.1 MPa) calculated in the DBA and DEC Chapter 15 increase in pressure events do not have test condition coverage from PANTHERS or other tests. It is possible that the outside of the condenser tubes could exceed the onset of critical heat flux (CHF) under those conditions and degrade the heat transfer performance of the ICS. If that were to happen, the Chapter 15 calculations would underestimate the peak pressure in the system. GEV provided information showing that CHF is conservatively modeled using Griffith's modification to the Zuber pool boiling model and that there is adequate margin to CHF in the DBA and DEC calculations in Chapter 15 where the pressures in the calculation exceed the pressures of the PANTHERS test data. A description of the Griffith's modification to the

Zuber CHF correlation and how it decreases the CHF value as the steam void fraction increases can be found in NUREG/CR-1559.

The staff makes the following findings with regards to meeting the applicable GDC and the proposed PDC:

The ICS system components are SC1 and seismic category 1 as shown in Table 3A-1 Preliminary BWRX-300 Component Classification List. The staff finds that the ICS will meet the requirements of GDC 2.

The ICS as described in the PSAR is designed to not be susceptible to instabilities and it will not have long horizontal pipe runs where pockets of steam trapped between regions of cold water could collapse and cause water hammer events. The staff finds that the ICS will meet the requirements of GDC 4

The ICS as described in the PSAR will not have components important to safety shared with other nuclear power units. The staff finds this will meet the requirements of GDC 5.

The PSAR proposes Principal Design Criterion (PDC) 17 as a substitute for GDC 17, but the ICS will not require onsite or offsite electrical power to function. The staff finds that the ICS as described in the PSAR can meet the requirements of PDC 17. The acceptability of PDC 17 is discussed in Section 8.1.3.

The reactivity control system has enough shutdown reactivity to compensate for the positive reactivity from the cold ICS water coming into the RPV. The staff finds that the ICS as described in the PSAR will meet the requirements of GDC 27.

The ICS as described in the PSAR has leak detection on the ICS steam supply and condensate return lines. The staff finds this will meet the requirements of GDC 32.

The ICS as described in the PSAR removes heat from the reactor core and minimizes inventory loss so that the core does not uncover and heat up in 72 hours. This is demonstrated in the ECCS calculations in PSAR chapter 15. The staff finds this will meet the requirements of GDC 35.

The ICS as described in the PSAR is designed to allow periodic inspection of the ICS components. The staff finds this will meet the requirements of GDC 36.

The ICS as described in the PSAR is designed to allow testing of the system with respect to integrity, performance, and operability. The staff finds this will meet the requirements of GDC 37.

The staff finds that the ICS as described in the PSAR will meet the requirements of 10 CFR 50.46a by providing a flow path from the high point of the ICS to the steam line that prevents the accumulation of non-condensable gas in the system during normal operation. The ICS condenser units have autocatalytic recombiners to prevent the accumulation of radiolytic non-condensable gases while the ICS units are operating. The autocatalytic recombiners are currently the subject of a research and development program.

The staff finds that the ICS as described in the PSAR will meet the requirements of TMI Action Plan item II.K.3.18 of NUREG-0737, equivalent to 10 CFR 50.34(f)(1)(vii), with respect to

eliminating the need for manual actuation of the BWR Automatic Depressurization System (ADS) to assure adequate core cooling. The BWRX-300 does not have an ADS system, but the ICS replaces the functionality and has an automatic actuation.

The staff finds that the ICS as described in the PSAR will meet the requirements of TMI Action Plan item III.D.1.1 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xxvi), with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the ECCS outside of the containment that contains or may contain radioactive material following an accident. The Process Radiation Monitoring Subsystem (PRM) monitors the IC pool area for radioactive materials that would indicate IC heat exchanger pressure boundary leakage as described in PSAR section 11.5.2.1.2.

6.3.3.4 Conclusion

The staff evaluated the ICS as described in PSAR section 6.3.3 against the regulatory requirements including applicable GDC, TMI Action Items, and 10 CFR 50.46a and determined that the design as described in the PSAR can meet the requirements for the portions of the design that are not deferred until the OL application. The autocatalytic recombiners are currently the subject of a development and testing program as described in the previous section. The design of the recombiners will need to be reviewed as part of the OL review to determine if they meet the required performance requirements for the design which is to keep the core from heating up for 72 hours as described earlier. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.3.4 Tests and Inspections (Isolation Condenser System)

6.3.4.1 Introduction

PSAR section 6.3.4 lists the requirements for testing and the proposed tests for the ICS in an operating power plant. The ICS must meet the requirements of ASME BPVC, Section III, Class 1 and 2 since it is part of the RPV pressure boundary and the ECCS system. The requirements specify that mandatory surveillance tests including In-Service Testing of components and in-service inspections (ISI) of pressure boundary components must be performed. The PSAR design allows the required testing including valve stroke testing and heat transfer performance testing to be performed. A startup test program for the ICS is described in Table 14.1-1 of the PSAR.

6.3.4.2 Conclusion

GDC 36 requires that the emergency core cooling system be designed to permit appropriate periodic inspections of important components. GDC 37 requires that the system be designed to permit appropriate periodic pressure and functional testing. The staff reviewed the tests and inspections that were proposed in PSAR section 6.3.4. The staff finds that the preliminary ICS design and proposed testing, as described in the PSAR, can meet the requirements of GDC 36 and 37 for periodic tests and inspections and is therefore acceptable. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

Commented [MMJ13]: Instead of this sentence, please add what portions we have reviewed and it meets the regulation?

Commented [GT14R13]: ~~the portions of the design that are not deferred until the OL application~~ this is already captured in the last sentence. Maybe. I am moving it back to the way it was for this sentence to make sense and fix your comment

6.3.5 Instrumentation Requirements (Isolation Condenser System)

6.3.5.1 Introduction

PSAR section 6.3.5 describes the instrument requirements to support actuation and isolation of the ICS system trains in the different modes of operation including ECCS, over-pressure protection, and shutdown cooling. The instrumentation must provide information to detect a break in the IC supply and return lines so they can be isolated in the event of a break. The instrumentation must also provide information that will show the operating status of the IC system when it is operating. The instrumentation includes flow and temperature instrumentation for the IC steam supply and condensate return lines, temperature instrumentation at multiple locations in the IC, and pressure instrumentation for the isolation valves nitrogen supply accumulators.

6.3.5.2 Conclusion

The staff reviewed the proposed instrumentation required to support the signal requirements for actuation, break detection, and isolation of the ICS system trains in the different modes of operation. The staff finds that the proposed instrumentation, which includes flow, temperature and pressure instrumentation and is adequate to support those functions. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.3.6 Conclusion

The applicant states that the design concept for emergency core cooling is based on preserving vessel inventory by removing decay heat from the RPV using the ICS and returning reactor coolant to the RPV. This reduces the pressure and the break flow in breaks that cannot be isolated. Removing decay heat also provides overpressure protection of the RCPB. Because of the volume of water in the RPV, there is no need to provide additional water from an external source to maintain core cooling in response to a postulated LOCA. 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. The ICS operates based on passive forces to provide overpressure protection and emergency core cooling once the system is on operation. The only components required to place an ICS train in operation are the parallel condensate return valves, each with the capacity for full flow of an ICS train. The condensate return valves fail open and are held open by positive means; the valves are redundant in each ICS train. The ICS system was designed to meet the required testing for ECCS systems and the ASME BPVC. A test program will be conducted to demonstrate the ICS autocatalytic recombiner functionality which is important for long term cooling. Startup testing requirements are given in PSAR Table 14.1-1. The applicant further states that the design conforms to the listed SRP sections and meets the listed regulatory requirements. The NRC staff finds that the information in PSAR Section 6.3 is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.34(a) and 10 CFR 50.35 as applicable.

Commented [MMJ15]: No regulation is cited for this section. Please add.

Commented [GT16R15]: The overall conclusion is just below. I have added a sentence with the 50.35 regulation.

Commented [MMJ17]: Is this the correct section number. 6.3.6 in the application is References and having this as the same number may cause confusion.

Is this supposed to be a full section conclusion that repeats the previous conclusions? If so, please delete it. This the only section thus far that has this style.

Commented [GT18R17]: Yes. It is the overall conclusion for Section 6.3. The formatting was changed during the PNNL review and was not properly linked to 6.3.5. I have fixed it. Regarding matching with the PSAR, that is not necessary. If you remember we had a huge issue in chapter 4.

6.4 Control Room Habitability

6.4.1 Introduction

This section identifies and describes the control room habitability design that will be used to support operator actions to operate the plant under normal and accident conditions. PSAR Section 6.4, "Control Room Habitability," discusses the plant design, which includes two distinct and independent control rooms; the main control room (MCR) and a secondary control room (SCR).

The MCR is located in the Control Building (CB) and has the capability to operate the plant during normal conditions as well as maintain the plant in a safe state and monitor critical parameters during an off-normal event. The SCR is located in the Reactor Building and provides the capability to initiate shutdown, maintain the plant in a safe shutdown condition, and monitor critical parameters during an off-normal event.

Control room habitability systems are those systems that support operator actions to operate the plant under normal and accident conditions. The CB heating, ventilation, and cooling system (HVS) design, which serves the MCR, is discussed in Section 9.4.1 of this SER. The Reactor Building HVS design, which serves the SCR, is discussed in Section 9.4.6 of this SER. For the BWRX-300 design, MCR and SCR habitability systems are not categorized as ESF systems because they are not associated with SC1 functions.

6.4.2 Regulatory Evaluation

In accordance with 10 CFR 50.34, "Contents of Applications; Technical Information", GDC 4, GDC 5, PDC 19, and 10 CFR 50.34(f), the control room must be designed to permit operator actions safely under normal conditions and to maintain it in a safe condition under accident conditions.

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

- In GDC 4, the NRC requires, in part, that systems for control room habitability shall be appropriately protected against dynamic effects and designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal operation, maintenance, testing, and postulated accidents.
- In GDC 5, the NRC requires, in part, that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.
- In GDC 19, the NRC requires, in part, that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions.
- The applicant is proposing to implement a design-specific PDC 19 in lieu of GDC 19 (evaluated in Section 7A & B.3 of this SER). The NRC staff reviewed the BWRX-300 control room design for consistency with PDC 19 as discussed below.
- In 10 CFR 50.34(f)(2)(xxviii), the NRC requires, in part, that in the event of an accident, radiation doses to operators do not exceed acceptable limits and do not prevent operators from performing control functions.

The following regulatory guidance provides acceptable methods that may be used to meet the relevant regulatory requirements:

- SRP Section 6.2.4, "Control Room Habitability System," provides criteria for meeting the relevant regulatory requirements as well as interfaces with other sections.
- Regulatory (RG) 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," provides guidance and criteria acceptable for compliance with NRC regulations regarding control room habitability.
- RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," provides guidance and criteria to consider for the impact of hazardous chemicals on control room operators.

6.4.3 Technical Evaluation

The NRC staff reviewed the CRN-1 preliminary design provisions for control room habitability in accordance with the guidance of SRP Section 6.4 and the requirements listed above. The results of the staff's review are provided below.

The control room habitability, as described in PSAR Section 6.4, is designed to maintain a suitable ambient temperature, quality of air, and pressurization for personnel and equipment in the MCR, the SCR and other areas of the MCR envelope during normal conditions and during an off-normal event. The MCR, which is located in the CB, is a Seismic Category II structure, while the SCR is located in the Reactor Building, which is a Seismic Category I structure.

In the event the MCR is no longer viable, operators can transfer control from the MCR to the SCR. An egress route exists from the MCR to the SCR for events that necessitate evacuation of the MCR. This may occur if the MCR's functionality is significantly impaired, if the room becomes uninhabitable, or if procedural criteria, such as an anticipated loss of MCR operability dictate relocation based on the nature of a specific event. For example, the SCR is the assured shutdown location for seismic events having a magnitude greater than the Design Basis Earthquake.

Because the SCR is located in a Seismic Category I structure, it is designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Therefore, the NRC staff concludes that the control room habitability design is consistent with GDC 4.

CRN-1 has only unit, and the control room habitability systems do not share any SSCs with other nuclear power plants. Therefore, the staff concludes that the requirements of 10 CFR Part 50, Appendix A, GDC 5 are met.

PDC 19 requires that nuclear plants operate safely under normal conditions, anticipated operational occurrences, and DBAs, while ensuring adequate radiation protection, human factors, and control room habitability (e.g., heating, ventilation, and air conditioning and lighting). The BWRX-300 design meets these requirements by maintaining the plant in a safe condition under DBA conditions without requiring operator actions from the MCR or SCR, and by limiting radiation exposure in the MCR or SCR to below 5 rem total effective dose equivalent during DBAs without credit for heating, ventilation, and cooling systems.

The control room habitability design uses RG 1.78 to ensure MCR habitability during hazardous chemical releases. For CRN-1, offsite chemical storage releases are not considered design basis events. However, mobile sources, such as traffic accidents involving chemicals like anhydrous ammonia, chlorine, hydrochloric acid, or nitric acid were evaluated. To protect MCR personnel, the CB HVS includes toxic gas detection at the outside air intake and automated operation of Toxic Gas Filtration Units. These units are backed by Standby Diesel Generator power during loss of normal air conditioning. This approach provides protection equivalent to RG 1.78, even though the CB HVS pressurization and isolation functions are classified as Safety Category 3 and Non-Seismic.

The design also includes remote safe shutdown capabilities outside the MCR, consistent with NRC guidance (e.g., SECY-94-084 and RG 1.189). In case of MCR evacuation, the reactor is tripped, and safety functions like decay heat removal and containment isolation are initiated before evacuation. Operators can then manage safe shutdown from the Seismic Category I SCR in the Reactor Building. This conforms to RG 1.196 by assuring that at least one control room is available to support fundamental safety functions for a DBA. Accordingly, the NRC staff finds that this meets the underlying purpose of the remote shutdown portion of GDC 19 by providing means for operators to maintain the reactor in a safe condition in the event of a control room evacuation.

The NRC staff finds that the control room habitability design ensures safe operation and shutdown under all conditions, including emergencies requiring MCR evacuation, and that the guidance of RG 1.196 and RG 1.78 has been followed. Therefore, the staff concludes that the BWRX-300 design complies with the requirements of PDC 19, which is consistent with GDC 19. Further discussion is located in section 7.A&B of this report.

To comply with 10 CFR 50.34(f)(2)(xxviii) for potential pathways for radioactivity and radiation that may lead to control room habitability problems, the applicant design description includes design provisions to preclude such problems. Specifically, for any operator actions claimed, including required monitoring of plant parameters, the design ensures that a habitable location is available and accessible during the associated event conditions for DBAs.

6.4.4 Conclusion

The NRC staff review concludes that the applicant's preliminary control room habitability design for CRN-1 is consistent with the NRC regulations as set forth in 10 CFR 50.34(f)(2)(xxviii) and Appendix A, GDC 4 and GDC 5, and with PDC 19, for a controlled environment for the comfort and safety of control room personnel and assures the operability of control room components during normal, anticipated operational transient, and DBA conditions because the applicant meets the acceptance criteria in SRP 6.4 and used acceptable methods in accordance with RG 1.196 for ventilation systems, and RG 1.78 for hazardous chemical releases.

The applicant states that the plant design includes two distinct and independent control facilities: an MCR and an SCR which assures that operator response can be accomplished for postulated initiating events. Control room habitability systems do not support Safety Category 1 functions. The applicant further states that the design conforms to the listed SRP sections and meets the listed regulatory requirements. Based on the design features, the staff concludes that the information provided to address control room habitability is acceptable for CP. Therefore, the NRC staff finds that the information in PSAR is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

Commented [MMJ19]: Acronym use is confusing CRN at time, CRN-1 at other, and it is not defined.

Commented [GT20R19]: All changed to CRN-1

Commented [MMJ21]: Normal operation?

Commented [MMJ22R21]: [@Cenarch Trusty](#) IS this supposed to be "norming operating"?

Commented [GT23R21]: Yes. Good catch, fixed.

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Feature Atmosphere Clean Systems

The applicant states that the design does not require an ESF atmospheric cleanup system. The BWRX-300 low-leakage containment is constructed in the subterranean, which is expected to further limit potential fission product leakage. The applicant further states that the design complies with the listed regulatory requirements. Based on the design features, the staff concludes that it is acceptable for CP.

6.5.2 Containment Spray Systems

The applicant states that this section is not applicable to the BWRX-300 design. Based on the design features, the staff concludes that it is acceptable for CP.

6.5.3 Fission Product Control Systems

The applicant states that the design does not use a dual or secondary containment. Containment isolation and leakage rate testing is addressed in Section 6.2. Based on the design features, the staff concludes that it is acceptable for CP.

6.5.4 Ice Condenser as a Fission Product Cleanup System

The applicant states that this section is not applicable to the BWRX-300 design. Based on the design features, the staff concludes that it is acceptable for CP.

6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System

The applicant states that This section is not applicable to the BWRX-300 design. Based on the design features, the staff concludes that it is acceptable for CP.

6.6 Preservice and In-service Inspection of Class 2 and Class 3 Components

6.6.1 Introduction

ASME Class 2 and 3 components should be designed to allow preservice inspections (PSI) and ISIs to be performed. Compliance with the regulations and GDCs identified below ensures that safety-system design will allow access to important components, so that periodic inspections can detect degradation, leakage, signs of mechanical or structural distress caused by aging and fatigue or corrosion before the systems are degraded to the point they cannot perform their intended safety functions.

6.6.2 Regulatory Evaluation

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

- 10 CFR 50.55a contains preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and 3 systems and components.
- GDC 36 requires that the design of the ICS permit appropriate periodic inspection of important safety components.

- GDC 37 requires that the design of the ICS permit appropriate testing to ensure structural integrity, leak tightness, and the operability of the system.
- GDC 39 requires that the design of the PCCS containment heat removal system permit inspection of important components, such as the torus and spray nozzles, to ensure the integrity and capability of the system.
- GDC 40 requires that the design of the PCCS containment heat removal system permit appropriate periodic pressure and functional testing to ensure the structural and leak-tight integrity of its components, the operability and performance of the active components of the system, and the operability of the system as a whole.
- GDC 42 requires that the design of the CIS permit appropriate periodic inspection of components such as filter frames and ducts to ensure integrity and capability of the system.
- GDC 43 requires that the design of the CIS containment atmospheric cleanup system permit appropriate periodic pressure and functional testing to ensure the structural integrity of components and the operability and performance of active components of the system, such as fans, filters, and dampers.
- GDC 45, "Inspection of cooling water system," requires that the design of the cooling water system such as the IC pools, equipment pool, and reactor cavity pool permit appropriate periodic inspection to ensure the integrity and capability of the system.
- GDC 46, "Testing of cooling water system," requires that the design of the cooling water system such as the IC pools, equipment pool, and reactor cavity pool permit appropriate pressure and functional testing to ensure the structural and leak-tight integrity of its components, the operability and performance of the active components of the system, and the operability of the system as a whole.

6.6.3 Technical Evaluation

The staff reviewed PSAR Section 6.6 of the TVA CPA PSAR, Revision 0, supplemented by letter dated January 7, 2026 (ML26007A164), against applicable regulatory requirements using regulatory guidance and standards to assess the sufficiency of the preliminary information on the PSI and ISI programs for the issuance of a CP in accordance with 10 CFR Part 50.

The staff's evaluation of the ISI program description of ASME Code Class 2 and 3 components is contained in the following seven sections that include (1) components subject to inspection, (2) accessibility, (3) examination categories and methods, (4) examination intervals, (5) evaluation of examination results, (6) system pressure tests, and (7) augmented ISI.

6.6.3.1.1 Components Subject to Inspection

PSAR Section 6.6.1 specifies that Quality Group B and C components (ASME Code Class 2 and 3, respectively) that are listed in PSAR Table 3A-1, "Preliminary BWRX-300 Component Classification List," are examined in accordance with ASME Code Section XI. The staff finds this acceptable because the applicant specified the Class 2 and 3 components to be inspected. However, the evaluation of the classification of ASME Code Class 2 and 3 components in accordance with the NRC quality group classification system (RG 1.26) or the definitions in Article NCA-2000 of Section III of the ASME Code is evaluated in Section 3.2 of this report.

6.6.3.1.2 Accessibility

PSAR Section 6.6.2 and Table 1.9-6, "Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)," specify that the ASME Code, Section III, Class 2 and 3 components are designed to provide access for the examinations required by ASME Code, Section XI, IWA-1500. In addition, PSAR Section 6.6.12 specifies that the BWRX-300 design meets the requirements for periodic inspection and testing of Class 2 and 3 components as specified in 10 CFR 50.55a and as detailed in ASME Code, Section XI. The staff notes that 10 CFR 50.55a(g)(2), "Accessibility requirements," provides the requirements for accessibility that components must be designed and provided with the access necessary to perform the required preservice and in-service examinations set forth in editions and addenda of Section III and Section XI of the ASME Code and 10 CFR 50.55a(g)(3)(ii), "Preservice examination requirements" provides the requirements that the Class 2 and 3 components must meet the PSI requirements set forth in the editions and addenda of Section III and Section XI of the ASME BPCV Code (in regards to access to both sides of the weld, removable insulation, etc.). The staff finds it acceptable that the applicant will base their ISI and PSI programs on the requirements of 10 CFR 50.55a and the ASME Code, which includes design of the components to provide access for inspections.

6.6.3.1.3 Examination Categories and Methods

PSAR Section 6.6.3.1 states that the PSI and ISI programs will follow ASME Code, Section XI, including IWC-2000 and IWD-2000, as required by 10 CFR 50.55a. Thus, the examination categories and methods specified are acceptable if they agree with the requirements in Articles IWA-2000, IWC-2000, and IWD-2000 of Section XI of the ASME Code. The staff will review the applicant's PSI and ISI programs prior to fuel load and prior to initial startup, respectively as defined in PSAR Table 13.4-1, "Operational Programs and Implementation Milestones."

In addition, PSAR Section 6.6.3.1 states that the examination category of each component using IWC-2000 and IWD-2000 of the ASME Code, Section XI will be listed in the PSI and ISI programs. Based on this, all components selected for in-service examination will receive a preservice examination, in accordance with ASME Code, Section XI, IWC-2200 and IWD-2200. The staff finds this acceptable since this all components will be included in the PSI program, which is consistent with 10 CFR 50.55a(g)(3)(ii), "Preservice examination requirements."

PSAR Section 6.6.3.2 specifies the examination methods are to be in accordance with ASME Code, Section V and Section XI. The examination methods include visual examinations, surface examinations (including eddy current testing, liquid penetrant testing, magnetic particle testing), volumetric examination (including eddy current testing, radiographic testing, ultrasonic testing), and alternative examination techniques that are newly developed which are demonstrated to be equivalent or superior to current examination methods. The NRC staff finds that the examination methods are consistent with ASME Code, Section XI, and that the alternative examination techniques must be demonstrated and approved by the NRC (as stated in PSAR Section 6.6.3.2) and therefore are acceptable methods. How the methods are specified in the PSI and ISI programs will be evaluated once the programs are developed based on the milestones in PSAR Table 13.4-1.

PSAR Section 6.6.3.2 specifies that the qualification of personnel and the examination equipment (systems) will be in accordance with ASME Code, Section V, Articles 2, 4, 5, 6, 7, 8, 9, 10, 12, and 13. However, ASME Code, Section XI has specific requirements for qualification of nondestructive examination (NDE) personnel and the NDE equipment and procedures, such

as IWA-2300, "Qualification of Nondestructive Examination Personnel," Mandatory Appendix I, "Ultrasonic Examinations," Mandatory Appendix III, "Ultrasonic Examination of Vessel and Piping Welds," Mandatory Appendix IV, "Eddy Current Examination," Mandatory Appendix VI, "Qualification of Personnel for Visual Examination," Mandatory Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic," and Mandatory Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems." In addition, NUREG-0800, Section 6.6 specifies that personnel, procedures and equipment are qualified to ASME Code, Section XI, including Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," and Appendix VIII, Performance Demonstration for Ultrasonic Examination Systems," to ASME Code, Section XI. In supplemental information provided by letter dated January 7, 2026 (ML26007A164), the applicant noted that NDE personnel, procedures and NDE equipment (systems) will be qualified in accordance with ASME Code, Section XI, including the Mandatory Appendices specified above because the PSI and ISI program is performed in accordance with ASME Code, Section XI. Ultrasonic examination systems will be qualified in accordance with an industry-accepted program for the implementation of ASME Code, Section XI, Appendix VIII. The staff finds this acceptable, given that any industry-accepted program is required to meet Appendix VIII requirements, in accordance with the implementation requirements of 10 CFR 50.55a.

6.6.3.1.4 Examination Intervals

The required examinations and pressure tests must be completed during each 10-year interval of service (hereafter designated as the inspection interval). In addition, scheduling of the program must comply with the provisions of ASME Code, Section XI, Article IWA-2000, concerning inspection intervals.

PSAR, Section 6.6.4, specifies inspection intervals for ASME Code Class 2 and 3 systems are consistent with ASME BPVC, Section XI, IWC-2400, and IWD-2400, respectively. Inspection intervals have a nominal length of 10 years with allowance for up to a year variation to coincide with refueling outages. The inspection intervals specified for ASME Code Class 2 and 3 components are consistent with the definitions in Section XI of the ASME Code and therefore are acceptable. In addition, PSAR Section 6.6 specifies that the initial ISI program will use the edition and addenda of ASME Code, Section XI, subject to the conditions identified and incorporated by reference into 10 CFR 50.55a, on the date no more than 18 months before the date of issuance of the operating license. The staff finds that the specific edition of the ASME Code, Section XI will be determined at the OL stage.

6.6.3.1.5 Evaluation of Examination Results

PSAR Section 6.6.6 specifies that examination results are evaluated in accordance with ASME Code, Section XI, Article IWC-3000, for ASME Code Class 2 components, with repairs based on the requirements of Article IWA-4000. Examination results are evaluated in accordance with ASME Code, Section XI, Article IWD-3000, for ASME Code Class 3 components, with repairs based on the requirements of Article IWA-4000. The description of the evaluation of examination results is consistent with ASME Code, Section XI, and meets the acceptance criteria in SRP Section 6.6, Section II.5, and is therefore acceptable.

6.6.3.1.6 System Pressure Tests

PSAR, Section 6.6.8, states that the system leakage and hydrostatic pressure tests requirements of IWA-5000 and IWC- 5000 will be met for ASME Code Class 2 components, and

the requirements of IWA-5000 and IWD-5000 will be met for ASME Code Class 3 components. The staff finds that this is acceptable since it is consistent with performing system leakage test and hydrostatic pressure tests in accordance with ASME Code, Section XI.

6.6.3.1.7 Augmented In-service Inspection to Protect against Postulated Piping Failure

The augmented ISI program for high-energy fluid-system piping, defined in PSAR Section 3.6 as the break exclusion zone, is acceptable if ISI examinations completed during each inspection interval provide a 100-percent volumetric examination of the pipe welds. The boundaries of the break exclusion zone run from the piping weld at the flanged connection to RIVs (inboard CIVs) to the outboard CIVs and past the outboard CIVs to the seismic interface restraints. The ICS does not have an outboard CIV, and all of the ICS piping is within the break exclusion zone.

PSAR Section 3.6.1.1 specifies the additional inspection requirements. A 100 percent volumetric in-service inspection is performed each inspection interval and the NDE techniques that have been demonstrated and qualified to detect threshold values of metal loss, cracking, or embrittlement as part of the break exclusion zone ISI procedure. The break exclusion zone piping is designed and laid out to provide access for ISI of every pipe segment, bend, and weld. The staff finds that the augmented inspection of the break exclusion welds to be acceptable since a 100 percent volumetric examination will be performed each inspection interval, which is consistent with BTP 3-4. The staff notes that these augmented inspections are an essential part of assuring that piping in the break exclusion zone is not subjected to leakage or rupture. The evaluation of the break exclusion zone is discussed in Section 3.6 of this SER.

6.6.3.1.8 Augmented Erosion/Corrosion Inspection Program

Bulletin 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," dated July 9, 1987, requested that operating reactor licensees submit information concerning their programs for monitoring the thickness of pipe walls in high-energy, single-phase and two-phase carbon steel piping systems. The staff subsequently issued GL 89-08, requiring operating reactor licensees to verify implementation of formalized procedures or administrative controls to ensure continued long-term implementation of the erosion and corrosion monitoring program for piping and components.

As described in GL 89-08, an appropriate long-term monitoring program for potential wall thinning of high-energy piping by erosion and corrosion must be implemented. The applicant has indicated that all piping systems that are ASME Code, Section III, Code ASME Code Class 1, 2, and 3, as well as nonsafety-related piping, and components described in GL 89-08 that are determined to be susceptible to erosion or corrosion shall be subject to NDE to verify system structural integrity. The applicant further stated that the examination schedule and methods shall be determined in accordance with Electric Power Research Institute (EPRI) guidelines in EPRI 3001000563, "Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R4)." The staff find this acceptable, because it meets current NRC guidance.

6.6.3.1.9 Augmented Inspection of Boiling Water Reactor Piping Susceptible to Intergranular Stress Corrosion Cracking

PSAR Section 6.6.10 specify that augmented inspections for intergranular stress corrosion cracking are performed in accordance with NUREG-0313, Revision 2, Generic Letter 88-01, and Supplement 1 to Generic Letter 88-01. Details of augmented inspections for intergranular stress corrosion cracking will be provided as part of the ISI program description in the FSAR. The staff finds that this is consistent with the NRC staff position on intergranular stress corrosion

cracking and will be evaluated once the PSI and ISI programs are developed and implemented based on the schedule in PSAR Table 13.4-1.

6.6.4 Conclusion

The staff concludes that the PSI and ISI programs for ASME Code Class 2 and 3 components that will be developed based on the schedule milestones in PSAR Table 13.4-1 is acceptable and will meet the inspection and pressure-testing requirements of 10 CFR 50.55a, as detailed in ASME Code, Section XI, and therefore satisfies the applicable requirements of GDC 36, 37, 39, 40, 42, 43, 45, and 46. The staff will review the applicant's PSI and ISI programs, once the PSI and ISI programs are developed based on the milestones in PSAR Section 6.6 and PSAR Table 13.4-1, "Operational Programs and Implementation Milestones." Therefore, the NRC staff finds that the information in PSAR Section 6.6 is sufficient and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.7 Main Steam Isolation Valve Leakage Control System

This section is not applicable to the BWRX-300 because the design does not include an MS isolation valve leakage control system. Based on the design features, the staff concludes that it is acceptable for CP.

6.8 Containment Monitoring Subsystem

6.8.1 Introduction

This section identifies and describes the CMon design that will be used to monitor certain containment parameters and notify plant personnel when measurements exceed preset limits. PSAR Section 6.8, "Containment Monitoring Subsystem," describes the instrumentation to monitor containment pressure, containment temperature, containment water level, containment hydrogen concentration, containment oxygen concentration, containment fission products, and containment area radiation levels.

6.8.2 Regulatory Evaluation

In accordance with 10 CFR 50.34; 10 CFR 20.1101(b); 10 CFR Part 20, Subpart F; 10 CFR 50.34(f)(2)(xvii); 10 CFR 50.34(f)(2)(xix); and GDC 64, the reactor containment must be designed to monitor certain parameters for hazardous conditions.

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

- In 10 CFR 20.1101(b), the NRC requires, in part, that sound radiation protection principles shall be used to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable.
- In 10 CFR Part 20, Subpart F, the NRC requires, in part, implementation of an effective survey and monitoring program to demonstrate compliance with NRC regulations.
- In GDC 64, the NRC requires, in part, that there shall be the capability provided to monitor the reactor containment atmosphere, for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.
- In 10 CFR 50.34(f)(2)(xvii), the NRC requires, in part, that instrumentation is provided to measure, record and readout in the control room for containment pressure, containment

water level, containment hydrogen concentration, containment radiation intensity (high level), and noble gas effluents at all potential, accident release points.

- In 10 CFR 50.34(f)(2)(xix), the NRC requires, in part, that instrumentation adequate for monitoring plant conditions is provided, following an accident that includes core damage.

The following regulatory guidance provides acceptable methods that may be used to meet the relevant regulatory requirements:

- RG 1.7, *Control of Combustible Gas Concentrations in Containment*, describes acceptable methods that may be used to comply with 10 CFR 50.44 for mitigation of combustible gas generated by a beyond DBA.
- RG 1.21, *Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste*, describes acceptable methods that may be used for measuring, evaluating, and reporting radiative material, and assessing and reporting public dose.
- RG 1.45, *Guidance on Monitoring and Responding to Reactor Coolant System Leakage*, describes acceptable methods that may be used for selecting reactor coolant leakage detection systems, monitoring for leakage, and responding to leakage.
- RG 1.97, *Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants*, describes an acceptable method that may be used to satisfy criteria for accident monitoring instrumentation.
- RG 8.2, *Administrative Practices in Radiation Surveys and Monitoring*, describes acceptable guidance in complying with the NRC's regulations for administrative practices associated with surveys and monitoring of ionizing radiation arising from licensed activities.
- RG 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable*, provides acceptable methods for meeting the criterion that exposures of station personnel to radiation during routine operation of the station will be as low as is reasonably achievable.

Technical Evaluation

- The NRC staff reviewed the CRN-1 preliminary design provisions for the containment monitoring subsystem in accordance with the guidance and requirements listed above. The results of the staff's review are provided below.
- Containment monitoring instrumentation includes pressure, temperature, and water-level transmitters. Pressure transmitters are installed outside the containment, with sensing lines routed through containment penetrations to monitor internal pressure. Temperature transmitters are distributed throughout the containment to track air temperature. Water level transmitters are strategically placed at various elevations to monitor water accumulation within the containment.
- Sample skids, located outside the containment, are used to monitor hydrogen and oxygen levels in the containment atmosphere. Samples are drawn through process piping routed via containment penetrations, analyzed, and then returned. Each skid is equipped with independent hydrogen and oxygen analyzers, flow control equipment, grab sampling taps, a local control panel, and audible/visual alarms. This conforms to RG 1.7 with regard to monitoring hydrogen and oxygen concentrations inside containment.
- A particulate, iodine, and noble gas skid is installed outside the containment to monitor fission products in the containment atmosphere. Process piping routes samples through

containment penetrations for analysis and returns them afterward. The skid features three independent analyzers (for particulates, iodine, and noble gases), flow control equipment, grab sampling taps, a local control panel, and audible/visual alarms. Filters and grab samples are collected periodically for isotopic analysis. This conforms to RG 1.21 with regard to monitoring for potentially hazardous contamination such as radioactivity, noble gases, air particulates, and halogens. Additionally, area radiation monitors are provided for monitoring gamma radiation levels in containment. The NRC staff finds that these instruments monitor the containment atmosphere for radioactivity releases during normal operations and off-normal events and therefore is consistent with GDC 64.

- The CMon provides instrumentation available for monitoring RCS leakage within containment such as containment temperature monitoring and containment fission product monitoring. The staff notes that an effective leakage monitoring strategy will include a combination of leakage monitoring instruments and methods. Monitoring changes in the following parameters can be useful for detecting a leak, as well as quantifying the flow rate. This conforms to RG 1.45 regarding monitoring parameters in support of RCPB leakage.
- Real-time measurements are displayed in the MCR, with alarms that alert personnel when values exceed predefined thresholds. Local indicators and alarms also are installed as needed to warn of hazardous conditions in specific areas. The NRC staff finds that the accident monitoring instrumentation, specifically containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity (high level), and noble gas effluents at all potential, accident release points, is consistent with 10 CFR 50.34(f)(2)(xvii) to measure, record and readout in the MCR.
- The CMon design for complying with 10 CFR 50.34(f)(2)(xix) is to conform to the guidance in RG 1.97 with regard to monitoring parameters in support of post-accident monitoring. A preliminary list of accident monitoring variables is provided in PSAR Table 7.3-2. The NRC staff finds the conformance of RG 1.97 acceptable for satisfying the requirements of 10 CFR 50.34(f)(2)(xix). A complete list of accident monitoring variables should be finalized in the FSAR.
- The CMon design for complying with 10 CFR 20.1101(b) and 10 CFR Part 20, Subpart F is to conform to the guidance in RG 8.2 and RG 8.8, respectively, with the use of engineering controls in maintaining doses as low as is reasonably achievable and by implementing a radiological monitoring program. The NRC staff finds the conformance of RG 8.2 and RG 8.8 acceptable for satisfying the requirements of 10 CFR 20.1101(b) and 10 CFR Part 20, Subpart F.

6.8.3 Conclusion

The NRC staff review concludes that the applicant's preliminary containment monitoring subsystem design for CRN-1 is consistent with the NRC regulations as set forth in 10 CFR 20.1101(b), 10 CFR Part 20, Subpart F, 10 CFR 50.34(f)(2)(xvii), 10 CFR 50.34(f)(2)(xix), and GDC 64, to monitor containment parameters for hazardous conditions because the applicant used acceptable methods in accordance with RG 8.2, RG 8.8, RG 1.7, RG 1.21, RG 1.45, RG 1.97.

The applicant stated that the CMon is a subsystem of the Process Radiation and Environmental Monitoring System and the design complies with the listed regulatory requirements. Based on the design features, the staff concludes that the information provided to describe the CMon is acceptable for CP. Therefore, the NRC staff finds that the information in the PSAR is sufficient

and can meet the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations of 10 CFR 50.35, as applicable.

6.9 References

ANSI/ANS-56.8, "Containment System Leakage Testing Requirements," American National Standards Institute/American Nuclear Society, 2020.

NEDC-33922P-A, "BWRX 300 Containment Evaluation Method," GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022 (ML22168A013).

NEDC-33911P-A "BWRX 300 Containment Performance," GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, January 2022 (ML22007A023).

NEDC-33926P-A, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design," GE-Hitachi Nuclear Energy Americas, LLC, Revision 4, December 2024 (ML25351A087).

ASME OM Code, "Operation and Maintenance of Nuclear Power Plants," American Society of Mechanical Engineers.

ASME BPVC, Section XI, Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," American Society of Mechanical Engineers. NEDC-33910P-A, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2021 (ML21183A261).

NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022 (ML22168A013).

EPRI 3001000563, "Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R4)," Electric Power Research Institute, 2013.