9 AUXILIARY SYSTEMS

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed Chapter 9 of the Tennessee Valley Authority's (TVA's or the applicant's) Construction Permit Application (CPA), Preliminary Safety Analysis Report (PSAR), as supplemented, against applicable regulatory requirements using regulatory guidance and standards to assess the sufficiency of the preliminary information on auxiliary systems for the issuance of a construction permit (CP) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 (TN249), "Domestic Licensing of Production and Utilization Facilities." As part of this review, the NRC staff evaluated information on auxiliary systems, with special attention given to design and operating characteristics, unusual or novel design features, and principal safety considerations. The NRC staff evaluated the preliminary design of the auxiliary systems to ensure the design criteria, design bases, and information relative to construction is sufficient to provide reasonable assurance that the final design will conform to the design basis for the one-unit BWRX-300 SMR (hereinafter also referred to as a CRN-1) designed by GE-Vernova Hitachi Nuclear Energy with a nominal electrical output of 300 MWe.

The NRC staff's reviews and evaluations for areas relevant to PSAR Chapter 9, including regulations and guidance used, a summary of the application information reviewed, and evaluation findings and conclusions, are discussed in the safety evaluation (SE) sections below for each specific review area.

9.1 <u>Fuel Storage and Handling</u>

9.1.1 New Fuel Storage

The CRN-1 proposed design does not include separate new fuel storage facilities; all new fuel is stored in the fuel pool area.

The NRC staff's evaluation of the new fuel storage is discussed in Section 9.1.2 of this report.

9.1.2 New and Spent Fuel Storage

9.1.2.1 Introduction

The spent fuel pool (SFP) provides onsite underwater storage of spent fuel assemblies and onsite underwater storage of new fuel assemblies. The SFP has the necessary design features unique to fuel storage during initial receipt, refueling operations, and accident conditions, including maintaining cooling and limiting offsite exposure in the event of a fuel handling accident. The fuel racks ensure that stored fuel is maintained in a suitable geometry to prevent criticality and provide cooling for all design conditions.

The applicant described the SFP structures, systems, and components (SSCs) related to fuel storage in PSAR Section 9.1.2. The applicant indicated that the CRN-1 proposed design does not include separate new fuel storage facilities, new fuel is stored in the same storage racks as the spent nuclear fuel. After receipt and inspection, the new fuel is placed in the fuel pool.

9.1.2.2 Regulatory Evaluation

This section summarizes the relevant regulations and the associated acceptance criteria as well as the review interfaces with other standard review plan (SRP) sections for this area of review:

- General Design Criteria (GDC) 2, "Design Bases for Protection against Natural Phenomena," as it relates to the capabilities of the SSCs important to safety to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.
- GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to capabilities of SSCs important to safety to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids.
- GDC 5, "Sharing of Structures, Systems, and Components," as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.
- GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to the requirement that the fuel storage system be designed to ensure adequate safety under normal and postulated accident conditions.
- GDC 63, "Monitoring Fuel and Waste Storage," as it relates to monitoring systems for detecting conditions that could cause the loss of residual heat removal capabilities for spent fuel assemblies, detecting excessive radiation levels, and initiating appropriate safety actions.
- 10 CFR 20.1101(b), as it relates to radiation doses kept as low as reasonably achievable (ALARA) (TN283).[1]
- 10 CFR 50.68 as to design to preclude criticality accidents (TN249).
- 10 CFR 50.34, "Contents of applications; technical information," including:

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide

[1] In accordance with 10 CFR 20.1002 (TN283), 10 CFR Part 20 does not apply to CPAs. 10 CFR 50.40 describes considerations that will guide the Commission in determining that a CP will be granted. Per 10 CFR 50.40(a), one of those considerations is that "the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations...including the regulations in part 20...and that health and safety of the public will not be endangered." Decisions related to the design of the proposed facility affect an applicant's ability to meet the requirements of 10 CFR Part 20 in the future. Therefore, discussions of 10 CFR Part 20 in this section pertain to the NRC staff having reasonable assurance that a future operating reactor would be able to meet 10 CFR Part 20 considering the preliminary facility design.

reasonable assurance that the final design will conform to the design bases with adequate margin for safety."

- 10 CFR 50.35, "Issuance of construction permits."

9.1.2.3 Technical Evaluation

The NRC staff reviewed PSAR Section 9.1.2, against the DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications," October 2022, and Chapter 9 of NUREG-0800 (NRC 2021-TN8013), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Revision 3, December 2016 to ensure that the PSAR includes the minimum design information relating to this review topic in accordance with 10 CFR 50.34(a), as applicable.

In PSAR Section 9.1.2.1, "Design Bases," the applicant identifies that the fuel pool performs Safety Category 1 Functions. The fuel pool provides safe, effective, and traceable storage of new and spent nuclear fuel. The fuel pool design also maintains and protects water in the fuel pool. This function is evaluated in Section 9.2.5, "Ultimate Heat Sink," of this SE because the fuel pool is part of the UHS.

NUREG-0800, SRP Section 9.1.2(III)(1), indicates that the minimum SFP storage capacity should equal or exceed the amount of spent nuclear fuel from 5 years of operation at full power plus one full-core discharge.

In PSAR Section 9.1.2.2 "Fuel Facilities Description," the applicant indicates that the storage pool will have capacity to store a batch of new fuel, a minimum of 5 years of spent nuclear fuel and a full core offload. The NRC staff notes that because the applicant proposes to combine new and spent nuclear fuel storage, its proposal is different from what the SRP contemplates. The SRP does not discuss minimum fresh fuel storage capacity, but even with the fresh fuel stored in the spent fuel pool, the spent fuel pool's capacity is consistent with the guidance in NUREG-0800, SRP Section 9.1.2(iii)(1), therefore the NRC staff finds this acceptable.

The NRC staff noticed that the PSAR does not provide the actual storage capacity but identified that the final design will meet the capacity identified in SRP Section 9.1.2(iii)(1). This level of detail is acceptable for a CPA because applicants are not required to submit more at the CP stage. The NRC staff will review the number of fuel storage locations, at the operating license (OL) phase, should the applicant apply for an OL.

NUREG-0800, SRP Section 9.1.2, provides the recommended design criteria for the proposed fuel storage. Among these criteria, the SRP indicates that the fuel racks should be designed so a fuel assembly can be inserted only in a design location, the fuel racks and any anchorages can withstand the maximum fuel handling equipment uplift forces without an increase in Keff or damage to the watertight integrity of the spent fuel pool liner, the racks provide adequate flow to prevent nucleate boiling, the fuel racks should be designed to protect the stored fuel from an safe shutdown earthquake (SSE).

PSAR Section 9.1.2.2, "Fuel Facilities Description," states that the fuel rack is designed to prevent accidental insertion of fuel assemblies between adjacent racks and allows flow to prevent nucleate boiling for all fuel assemblies. The fuel racks are also designed to withstand the effect of a stuck fuel assembly. The fuel racks are designed to withstand an SSE without failure of the basic structure or damage to the active region of irradiated fuel.

The spent nuclear fuel storage racks are designed with neutron absorbing materials, the NRC staff evaluation of all criticality-related concerns, including whether the fuel racks and any anchorages can withstand the maximum fuel handling equipment uplift forces without an increase in Keff or damage to the watertight integrity of the spent fuel pool line, in the fuel pool are discussed in Section 9.1.A of this report.

The NRC staff finds that the applicant has identified adequate design criteria for the fuel storage rack that are consistent with the guidance discussed in SRP 9.1.2.

GDC 2, "Design Bases for Protection against Natural Phenomena"

GDC 2 requires, in part, that nuclear power plant SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquake, tornado, hurricane, flood, tsunami, and seiche without loss of capability to perform their safety functions. The design bases of these SSCs also must reflect appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena.

The functions of the new and spent fuel storage facilities are to maintain new and spent nuclear fuel in a safe and subcritical array during all anticipated operating and accident conditions and to limit offsite exposures in the event of significant release of radioactive materials from the fuel. Further, GDC 61, which is discussed below, required that the spent nuclear fuel storage facility also must keep spent fuel assemblies adequately cooled during all anticipated operating and accident conditions.

GDC 2 assures that SSCs of the new and spent fuel storage facilities (e.g., the spent fuel pool, new and spent nuclear fuel storage racks, and pool liner) will withstand the effects of natural phenomena that might occur at the plant site.

GDC 2 provides assurance that natural phenomena will not prevent maintenance of a subcritical configuration and adequate cooling of the stored fuel.

In PSAR 9.1.2.3, "Safety Evaluation," the applicant indicates that the fuel pool liner and fuel storage racks are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, and floods without loss of capability to perform their safety functions. They meet Seismic Category I requirements and are protected from interaction with non-Seismic Category I SSCs to ensure that the SSE does not cause a loss of capability to perform their safety functions. In PSAR Section 9.1.2.2 "Fuel Facilities Description," the applicant indicates that the storage racks are located in the deep pit of the fuel pool, which is constructed of a steel and concrete composite. The fuel pool is an integral part of the Reactor Building (RB) structure and designed to meet Seismic Category I requirements.

The NRC staff finds that designing the fuel pool and fuel storage racks to meet seismic Category I requirements would assure that earthquakes will not cause a substantial coolant loss, a reduction in margin to criticality, or damage to the fuel assemblies. The PSAR identified the design criteria for the fuel racks. The NRC staff noted that the PSAR does not include an analysis that demonstrates that the rack meets the regulatory requirements. However, complete design information is not required under NRC regulations to support issuance of the CP. Nothing in the regulations requires this analysis at the CP stage. Thus, at the final safety analysis report (FSAR) review stage, should the applicant apply for an OL, the NRC staff will evaluate the FSAR and the thermal analysis of the spent fuel pool (and Ultimate Heat Sink, for some accident scenarios) to confirm that the final design of the new and spent nuclear fuel storage facility meets the applicable design requirements.

The NRC staff finds that the PSAR has identified relevant requirements and provided sufficient description of preliminary design criteria of the fuel racks to authorize construction. Therefore, the NRC staff finds that the design criteria presented in the PSAR are consistent with the requirements of GDC 2 and support issuance of the CP.

GDC 4, "Environmental and Dynamic Effects Design Bases"

Compliance with GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and be compatible with, the environmental conditions of anticipated normal operating and postulated accident conditions. This requirement includes protection against dynamic effects, including those of missiles, pipe whipping, and discharging fluids caused by equipment failures and from events and conditions outside the nuclear power unit.

In PSAR Section 9.1.2.3, "Safety Evaluation," states that the fuel pool liner and fuel storage racks are design to comply with GDC 2 and 4, and conform to Regulatory Guide (RG) 1.13, "Spent Fuel Storage Facility Design Basis," Rev. 2, positions C.1 and C.2. They are designed to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accident (LOCA) and the dynamic effects of equipment failure. The PSAR also indicates that the liner and fuel storage racks are designed to meet Seismic Category I requirements and are protected from interaction with non-Seismic Category I SSCs to ensure that the SSE does not cause a loss of capability to perform their safety functions. The PSAR also indicates that the fuel pool liner and storage racks are protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluid, which may result from equipment failures and events and conditions outside the nuclear power unit.

The NRC staff noted that the storage racks are located in the deep pit in the fuel pool. The NRC staff finds that the thick seismic Category I walls of the pool and the water above the racks provide the necessary protection from dynamic effects for SSCs within the pool. Therefore, the NRC staff finds that the applicant has provided adequate design criteria and design information related to the fuel storage system such that the NRC staff has reasonable assurance that the fuel storage system can be constructed to meet the requirements of GDC 4. Thus, the NRC staff finds that the design criteria presented in the PSAR is consistent with the requirements of GDC 4 at the CP stage. Therefore, the NRC staff finds that the PSAR adequately addresses fuel pool design is in accordance with 10 CFR 50.34(a)(3)(ii).

GDC 5, "Sharing of Structures, Systems, and Components"

GDC 5 requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the fuel pool design can be in conformance with GDC 5.

GDC 61, “Fuel Storage and Handling and Radioactivity Control”

GDC 61 requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. The fuel storage system must be designed with (1) the capability for appropriate periodic inspection and testing of components important to safety, (2) suitable shielding for radiation protection, (3) appropriate containment, confinement, and filtering capability, (4) residual heat removal that reflects the safety importance of decay heat and other residual heat removal, and (5) the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 1—Periodic Inspection and Testing

In PSAR Section 9.1.2.3 the applicant indicated that the fuel storage racks are designed to allow inspection for the presence of neutron poison. The fuel pool is an integral part of the RB and therefore covered under the in-service inspection (ISI) and testing program that is established for the RB. PSAR Section 9.1.2.4, “Inspection and Testing,” states that the design of the fuel pool liner and fuel storage racks facilitates inspections and testing to verify that there is no corrosion of the fuel pool liner or fuel storage racks, no buildup of crud or debris that may obstruct coolant flow in the fuel pool, and no degradation of any strong fixed neutron absorbers.

The NRC staff evaluated the fuel pool inspection capability described in the PSAR and discussed above and finds that the applicant has designed the fuel pool with the capability for appropriate periodic inspection and testing of components important to safety, in accordance with GDC 61.

Criterion 2—Shielding

The PSAR indicates that the fuel pool is designed to maintain sufficient water above the top of the stored fuel and the fuel in transfer to provide adequate radiation shielding. The pool nominal water level maintains a nominal 23 ft of water above the top of the active fuel stored in the racks. The fuel pool is part of the ultimate heat sink and connected with other pools that could drain water out of the fuel pool. The design includes a gate and weir that prevents drainage of coolant inventory below adequate shielding depth. Additionally, the fuel pool does not have any drain, pipping, or other installed system that would drain the pool water level below adequate shielding depth. The PSAR indicates that during plant operation the gates are in place, in this configuration, the design conforms to position C.6 of RG 1.13 (NRC 2007-TN13096).

SRP Section 9.1.2.III.2.H.i states that the SFP design should include weirs and gates separating the spent nuclear fuel storage areas from handling areas to prevent the accidental draining of the coolant to levels inadequate for fuel cooling or radiation shielding. The bottom of any of the gates should be above the top of the fuel assemblies, and the volume of the adjacent fuel handling areas should be limited so that leakage into these areas while drained would not reduce the coolant inventory to less than 3 m (10 ft) above the top of the fuel assemblies.

The NRC staff evaluated the system configuration described in the PSAR and found that the bottom of the fuel pool gate is located above the top of the fuel assemblies. PSAR Section 9.1.2 indicates that the fuel pool gate is provided with two radiation resistant seals in series. Table 3A-1 identifies the fuel pool gate and the equipment pool gate as safety class 1 SSCs design to seismic Category 1 standards.

Based on the system configuration described in the PSAR, the NRC staff finds that the proposed location of the gates and their seismic design are acceptable to ensure the coolant inventory is maintained at or above 3 m above the top of the fuel assemblies. This is consistent with the guidance in SRP Section 9.1.2.III.2.H.i.

The application system description states that sufficient water depth is maintained in the fuel pool to provide adequate scrubbing of fission gases during a postulated fuel handling accident. The NRC staff evaluation of the fuel handling accident is evaluated in Section 15.5.7 of this SE.

The NRC staff noted that the PSAR has not indicated the actual minimum water level that will be maintained in the fuel pool credited for the scrubbing of fission gases during a postulated fuel handling accident. This is acceptable because complete design information is not required under the NRC regulations to support issuance of the CP. At the time of FSAR review, should the applicant apply for an OL, the NRC staff will determine whether the minimal water level maintained in the pool is sufficient to provide adequate shielding.

The fuel pool is provided with level instrumentation to monitor the fuel pool water level, ensuring that the minimum water level is maintained. The instrumentation design criteria are discussed and evaluated in Section 9.1.3.

The NRC staff evaluated the fuel pool design criteria discussed above and finds that the applicant has identified adequate design criteria to demonstrate that the fuel pool constructed in accordance with these criteria will provide adequate shielding for radiation protection. Thus, the information provided is consistent with this criterion of GDC 61.

<u>Criterion 3—Containment, Confinement, and Filtering</u>

In the PSAR the applicant indicates that the fuel pool liner, fuel pool, and the walls of the RB integral with the fuel pool provide containment of the fuel assemblies and the water cooling the assemblies. Radionuclides released into the water are collected and filtered by the fuel pool cooling and cleanup system (FPC), which is described in PSAR Section 9.1.3. The NRC staff evaluation of the FPC is discussed in Section 9.1.3.

The fuel pool is located in the RB, which is serviced by a ventilation system. The RB heating, ventilation, and cooling system is discussed in PSAR Section 9.4.6. The RB ventilation system provides containment and confinement to limit releases of airborne radioactivity to the environment from the fuel pool under normal and postulated accident conditions. The NRC staff's evaluation of the RB heating, ventilation, and cooling system is presented in Section 9.4.6.

The NRC staff finds that crediting the water contained in the fuel pool, FPC, and the RB heating, ventilation, and cooling system would provide adequate design criteria for ensuring containment, confinement, and filtering of radionuclide in the fuel pool, as required by GDC 61.

<u>Criterion 4—Residual Heat Removal</u>

The PSAR indicates that the normal decay heat removal function is performed by the FPC, which is described in PSAR Section 9.1.3, and it is not credited to function during or following an accident scenario. The NRC staff evaluation of the FPC is discussed in Section 9.1.3 of this report.

For decay heat removal during or following an accident scenario, the fuel pool credits the volume of water retained in the pool to remove decay heat. The PSAR indicates that the pool walls, pool liner, pipping connections and gates are designed to prevent pool draindown that could impact decay heat removal capability. PSAR Section 9.1.2 indicates that the pool contains sufficient water inventory to allow 7 days of passive spent fuel decay heat removal without crediting makeup, while maintaining sufficient coverage of 10 ft over fuel.

The NRC staff noted that the PSAR does not identify the minimum water level (or volume) needed to provide decay heat removal capability capable of providing 7 days of passive spent fuel decay heat removal without crediting makeup, while maintaining sufficient coverage of 10 ft over fuel. This level of detail is sufficient because complete design information is not required under NRC regulations to support issuance of the CP. At the FSAR review stage, should the applicant apply for an OL, the NRC staff will confirm that the spent fuel pool thermal analysis demonstrates that the minimum water level maintained in the spent fuel pool is adequate to allow 7 days of passive spent fuel decay heat removal without crediting makeup, while maintaining sufficient coverage of 10 ft over fuel. The NRC staff will also confirm that the FSAR identifies the pool parameters monitored and control to ensure the pool conditions are bounded by the spent fuel pool thermal analysis.

The NRC staff evaluation of the fuel cooling design criteria is discussed in Section 9.2.5. The NRC staff finds that the design features discussed above are adequate to ensure the fuel pool retains adequate water inventory to provide the required residual heat removal identified in GDC 61.

<u>Criterion 5—Drainage Prevention</u>

PSAR Section 9.1.2.3 indicates that when the fuel pool gate is installed, separating the fuel pool from the reactor cavity pool, the fuel pool design is in conformance with position C.6 of RG 1.13, since there are no drains, piping, or other systems installed in the fuel pool that would allow coolant levels to drain below adequate shielding depths of above the top of the fuel assemblies. The PSAR also indicates that the pool liner is provided with leakage monitoring to ensure liner integrity.

During outages, the fuel pool gate may be removed, the fuel pool is connected to the volume of water in the reactor cavity pool. When the pools are connected, the reactor cavity pool maintains water above the minimum level credited for shielding. Reactor cavity integrity is evaluated in Section 9.2.5.

Based on the discussed above, the NRC staff evaluated the fuel pool design features credited to prevent pool drainage and finds that these design features are consistent with the guidance in SRP 9.1.2. As such, and based on the discussion of the criterion above, the NRC staff finds that the applicant has identified adequate design criteria for the new and spent nuclear fuel storage facility that conforms with the requirements of GDC 61.

The NRC staff verifies that the materials wetted in the spent fuel pool, (e.g., spent fuel racks, fixed neutron poison, and the spent fuel pool liner) are chemically compatible and stable. The review also verifies whether there are potential mechanisms to alter the dispersion of any strong fixed neutron absorbers. These principles are evaluated in Section 3.8 of this report.

GDC 63, “Monitoring Fuel and Waste Storage”

GDC 63 requires appropriate systems for fuel storage, radioactive waste, and handling areas to detect conditions that may result in a loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions.

For spent fuel storage facilities, GDC 63 requires monitoring of the SFP water level, pool temperature, and pool building radiation levels to protect personnel, prevent significant offsite radiation doses, and detect conditions that could cause the loss of decay heat removal capabilities. In addition, alarms and communications systems should alert personnel and allow communication between fuel handling machine (FHM), refueling machines and the control room (CR). If necessary, to limit offsite dose consequences from a fuel handling accident or pool boiling, instrumentation should automatically place the spent nuclear fuel facility ventilation system in a mode to reduce the offsite release of radioactive material.

In PSAR Section 9.1.2.3 the applicant indicates that the fuel pool will be provided with instrumentation to detect conditions that may cause loss of residual heat removal capability and excessive radiation levels, and to initiate appropriate actions. The applicant stated that this is done consistent with RG 1.13 position C.7, and paragraph 5.4 of ANSI/ANS-57.2, which is endorsed by the NRC in RG 1.13. The PSAR also indicates that instrumentation provides data for control functions and for local information and alarms.

Instrumentation for identifying fuel pool water level, pool temperature, and radiation monitoring is part of the FPC and discussed in PSAR Section 9.1.3. The NRC staff evaluation of the FPC instrumentation is discussed in Section 9.1.3.

The NRC staff finds that the description of the fuel pool instrumentation design criteria consistent with the guidance provided in SRP 9.1.2. Therefore, the NRC staff finds that the applicant has identified adequate design criteria for instrumentation capable of detecting and monitoring conditions that may result in a loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions, in conformance with GDC 63.

As Low as Reasonably Achievable Principle

Compliance with 10 CFR 20.1101(b) requires the licensee to use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to achieve ALARA occupational doses and doses to the public. As noted above, however, regulations in 10 CFR Part 20 do not apply at the CP phase. As such, the NRC staff’s review is limited to determining whether there is reasonable assurance the regulations will be met, consistent with 10 CFR 50.40(a).

In PSAR Section 9.1.2.3, the applicant indicates that it is following the guidance provided in RG 1.13 and ANSI/ANS-57.2 paragraph 5.1.5, for maintaining safe radiation levels for personnel during anticipated operating and accident conditions. Additional information on the design of SSCs relative to ALARA is presented in PSAR Section 12.3. The NRC staff evaluation of this information is discussed in Section 12.3.

9.1.2.4 Conclusion

The NRC staff evaluated the fuel pool system design description provided in PSAR 9.1.2 and concluded that the applicant has proposed adequate design criteria for the new and spent nuclear fuel storage system. The NRC staff has confidence that a fuel pool system constructed in accordance with these criteria could meet the requirements of 10 CFR 20.1101(b) and GDC 2, 4, 5, 61, and 63 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

The applicant will complete the final fuel storage rack design and the thermal analysis of the spent fuel pool to demonstrate that the facility meets the applicable requirements of GDC 2, 4, 5, 61, and 63 of Appendix A of 10 CFR Part 50, 10 CFR 50.68 or 10 CFR 70.24 and make it available for staff review and approval as part of an OL application (OLA) at the CRN Site.

9.1.3 Fuel Pool Cooling and Cleanup System

9.1.3.1 Introduction

All nuclear reactor plants include an SFP for the wet storage of spent fuel assemblies. The methods used to provide cooling for the removal of decay heat from the stored assemblies vary from plant to plant, depending upon the individual design. The safety function to be performed by the system in all cases remains the same; that is, the spent fuel assemblies must be cooled and must remain covered with water during all storage conditions. The NRC staff's review, documented in this section is to assure adequate cooling of spent fuel, this includes both the active FPC during normal operation and maintaining sufficient water inventory to allow for passive cooling through the boiling of spent fuel pool water.

PSAR Section 9.1.3, discusses the design and performance of the FPC. The FPC is designed to provide cooling of the water volume in the fuel pool, to provide makeup coolant inventory to the fuel pool, to ensure spent nuclear fuel is kept cool and submerged until relocated for permanent storage, to maintain water quality and to reduce general area dose. The FPC can also be aligned to provide cooling and cleanup to the reactor cavity and equipment pools, as necessary.

During accident scenarios, the proposed design credits the water inventory stored in the fuel pool/UHS to passively remove the decay heat from the stored fuel in the fuel pool. The NRC staff evaluates the UHS in Section 9.2.5 of this SER.

9.1.3.2 Regulatory Evaluation

The relevant regulations for this area of review and the associated acceptance criteria, as well as the review interfaces with other SRP sections, are:

- GDC 2, as it relates to SSCs important to safety being designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and seiches. GDC 2 is not applicable to the cleanup portion of the system and need not apply to the cooling system if both the fuel pool makeup water system (and its source) and the auxiliary building (and its ventilation and filtration system) meet this criterion.
- GDC 4, as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs and dynamic effects resulting from pipe whip, missiles, and discharging fluids.

- GDC 5, as it relates to shared SSCs important to safety not being shared among nuclear power units, unless it can be shown that such sharing will not significantly impair performing required safety functions.
- GDC 61, as it relates to the requirement that the fuel storage system be designed to assure adequate safety under normal and postulated accident conditions, including the capability to permit appropriate periodic inspection and testing of components important to safety; suitable shielding for radiation protection; appropriate containment, confinement, and filtering capability; residual heat removal capability that reflects the importance to safety of decay heat and other residual heat removal; and the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.
- GDC 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capability, to detect excessive radiation levels, and to initiate appropriate safety actions.
- CFR 20.1101, as it relates to radiation doses being kept ALARA (TN283).[2]
- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.1.3.3 Technical Evaluation

PSAR Section 9.1.3.1 identifies that the Safety Category 1 function of protecting the water required for cooling, shielding, and subcriticality control of the fuel in the fuel pool is discussed in PSAR Section 9.1.2, and the NRC staff evaluation is discussed in Section 9.1.2 of this report. The Safety Category 1 function of retaining water in the UHS pools for passive containment cooling system (PCCS) is discussed in PSAR Section 9.2.5, and the NRC staff evaluation is discussed in Section 9.2.5 of this report.

The applicant indicated that the FPC has no Safety Category 2 Functions. The system Safety Category 3 Functions are:

[2] In accordance with 10 CFR 20.1002, 10 CFR Part 20 does not apply to CPAs. 10 CFR 50.40 describes considerations that will guide the Commission in determining that a CP will be granted. Per 10 CFR 50.40(a), one of those considerations is that "the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations...including the regulations in part 20...and that health and safety of the public will not be endangered." Decisions related to the design of the proposed facility affect an applicant's ability to meet the requirements of 10 CFR Part 20 in the future. Therefore, discussions of 10 CFR Part 20 in this section pertain to the NRC staff having reasonable assurance that a future operating reactor would be able to meet 10 CFR Part 20 considering the preliminary facility design.

- cooling of the fuel pool, reactor cavity pool, and equipment pool
- makeup water capabilities for the fuel pool, reactor cavity pool, and equipment pool, delivered directly to the surge tanks and circulated to the pools
- maintaining the water level of the fuel pool for shielding and cooling
- direct makeup water capabilities for the fuel pool during accident conditions, independent of the forced cooling portion of the system
- maintaining the water level of the fuel pool for shielding and cooling during accident conditions

The system Safety Category N Functions include the following:

- maintaining the water quality of the fuel pool, reactor cavity pool, and equipment pool through filtration and demineralization
- restoring the fuel pool temperature to normal operating limits from elevated temperatures, caused by a system outage or accident conditions, upon restoration of the forced cooling components of the system.

GDC 2, "Design Bases for Protection against Natural Phenomena"

Compliance with GDC 2 requires that SSCs important to safety be designed to withstand the effects of expected natural phenomena combined with the appropriate effects of normal and accident conditions without a loss of capability to perform their safety functions. SRP 9.1.3 states that conformance to RG 1.13, rev 2 (NRC 2007-TN13096), Regulatory Positions C.1, C.2, C.6, and C.8; and RG 1.29, Regulatory Position C.1, for safety-related portions of the system, and Regulatory Position C.1.i, for portions of the system that are not safety related are acceptable acceptance criteria for meeting GDC 2.

RG 1.13, Regulatory Position C.1, states that the spent fuel storage facility, including all structures and equipment necessary to maintain the minimum water levels needed for radiation shielding, should be designed to meet seismic Category I requirements. RG 1.13, Regulatory Position C.2, states that the spent nuclear fuel storage facility should be designed to (1) keep extreme winds and missiles generated by those winds from causing significant loss of watertight integrity of the fuel storage pool, and (2) keep missiles generated by extreme winds from contacting fuel within the pool.

The Safety Category 1 functions identified in PSAR Section 9.1.3 are discussed in Section 9.1.2 and 9.2.5 of the PSAR and evaluated in Section 9.1.2 and 9.2.5 of this report. The FPC has no function identified as important to safety.

RG 1.13, Regulatory Position C.6, "Drainage Prevention," states that the drains, permanently connected mechanical or hydraulic systems, and other features that (by maloperation or failure) could reduce the coolant inventory to unsafe levels should not be installed or included in the design.

PSAR Section 9.1.2 discusses fuel pool drainage prevention, and the NRC staff evaluated this in Section 9.1.2 of this report. In PSAR Section 9.1.3.5, the applicant indicates that the FPC has no drains, piping, or other systems that would allow fuel pool water to drain below the minimum level needed to support plant safety analyses, which is above the level needed for adequate shielding of the fuel assemblies.

As stated before, the NRC staff has noted that the applicant has not identified the specific minimum water level that is credited to support plant safety analyses. As stated above, this is acceptable because complete design information is not required under the NRC regulations to support the issuance of the CP. At the time of FSAR review, the NRC staff will confirm that the FPC drains, piping, or other systems components are located to prevent drain down of the fuel pool below the minimum level needed to support plant safety analyses. The NRC staff finds that the FPC design criteria are consistent with the guidance in RG 1.13, Regulatory Position C.6.

RG 1.13, Regulatory Position C.8, “Makeup Water,” states that a Quality Group C, seismic Category I makeup system should be provided to add coolant to the pool. PSAR Section 9.2.5 states that the fuel pool is designed to retain enough water inventory following any accident scenario to allow for 7 days of passive pool cooling without crediting additional makeup. In PSAR Section 9.1.3.3.2, the applicant indicated that the FPC is designed with the capability of providing makeup water to the fuel pool from multiple water sources. The PSAR also indicates that a seismic Category I connection is provided from the outside of the RB into the fuel pool. This connection allows for the use of FLEX/Emergency Mitigating Equipment (EME) to provide make up water to the pool within 7 days of the normal makeup sources’ unavailability.

As stated above, the NRC staff evaluated the proposed design of the fuel pool makeup sources and noted that the fuel pool is designed as a passive pool with sufficient water inventory to provide adequate cooling for 7 days without crediting makeup. The design includes multiple active makeup water sources to the fuel pool. Based on the FPC makeup capabilities discussed above, the NRC staff finds that the applicant has identified adequate design criteria that are consistent with the guidance of RG 1.13, Regulatory Position C.8. Additionally, the design includes provision to use the FLEX/EME as an additional makeup water source. The NRC staff will evaluate FLEX/EME and its capabilities as part of the review of the FSAR when the design is finalized because this evaluation is not required under NRC regulations to support issuance of the CP.

The NRC staff’s evaluation of how the fuel pool and the UHS design criteria meet the requirements of GDC 2 are discussed in Section 9.1.2 and 9.2.5 this report. The NRC staff evaluated the design of the FPC presented in PSAR Section 9.1.3 and found that the applicant has identified adequate design criteria consistent with the guidance provided in the applicable portions of RG 1.13, RG 1.29. As such, the design criteria are consistent with GDC 2.

GDC 4, “Environmental and Dynamic Effects Design Bases”

Compliance with GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operating and postulated accident conditions. This requirement includes protection against dynamic effects, including those of missiles, pipe whipping, and discharging fluids caused by equipment failures and from events and conditions outside the nuclear power unit.

In PSAR Section 3.1.1.4 the applicant discusses how the design criteria meet the requirements of GDC 4. PSAR Section 3.1.1.4 addresses all systems, not just the FPC. The PSAR indicates that SSCs classified as Safety Category 1 are designed to be compatible with environmental conditions associated with normal operations, maintenance, testing, and postulated pipe failures accidents.

The SSCs associated with the Safety Category 1 Functions are evaluated in Sections 9.1.5 and 9.2.5 of this report. The NRC staff's evaluation of adequate missile protection of Safety Category 1 SSCs is discussed in Section 3.5 of this report.

The NRC staff reviewed the system description of the FPC and identified that the remaining components of the FPC have no Safety Category 1 function that requires protection. When the FPC is not able to perform its cooling function, the proposed design relies on the water inventory retained in the fuel pool to cool the stored fuel.

The NRC staff evaluated the FPC design discussed in PSAR Section 9.1.3 and based on the reviews at the above-cross referenced sections, identified that the applicant has adequately identified design criteria that would ensure the final design of the FPC complies with the requirements of GDC 4.

GDC 5, "Sharing of structures, systems, and components"

GDC 5 requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units.

The CRN-1 Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the fuel pool design can be in conformance with GDC 5.

GDC 61—Fuel storage and handling and radioactivity control, as it relates to handling of radioactive materials

GDC 61 requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. The fuel storage system must be designed with (1) the capability for appropriate periodic inspection and testing of components important to safety, (2) suitable shielding for radiation protection, (3) appropriate containment, confinement, and filtering capability, (4) residual heat removal that reflects the safety importance of decay heat and other residual heat removal, and (5) the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

In SRP Section 9.1.3, Rev. 2, the NRC staff identify the design provisions that should be credited to demonstrate compliance with GDC 61. One of these provisions states that the requirement for the design of the spent fuel pool cooling and cleanup system to include provisions for residual heat removal that reflect its importance to safety is satisfied by (1) designing essential portions of the cooling system to seismic Category I criteria and with adequate cooling capacity assuming a single active failure, and (2) providing a forced-circulation cooling capability that maintains the pool at temperatures suitable for fuel handling during routine operating conditions, including refueling.

PSAR 9.1.3.3.1 states that the FPC is designed to actively remove decay heat from the fuel pool and maintain the fuel pool water below 140°F, including while containing a full core offload during refueling operations. The PSAR Section 9.1.3.2 indicates that one of the two system trains has sufficient decay heat removal capacity to prevent bulk boiling of the water in the fuel pool under maximum expected fuel pool heat load conditions. When the FPC is not available, the fuel pool is designed to retain sufficient water level to remove decay heat for 7 days without crediting makeup water. When makeup water is needed, the FPC system is designed to provide makeup from multiple sources. The FLEX/EME strategy is credited with providing makeup to the fuel pool when the FPC is not available.

The NRC staff finds that designing the FPC with sufficient capacity to maintain the fuel pool water below 140°F for all anticipated operations is consistent with the guidance of SRP 9.1.3.1.H. An FPC system design with this capacity would provide adequate SFP cooling capacity for routine operations, including refueling.

The NRC staff noted that, although the application does not include a pool thermal analysis that identifies heat loads or cooling requirements, complete design information is not required under the NRC regulation to support issuance of the CP. The final thermal analysis will be reviewed at the OL review stage, should they applicant submit an OLA, to confirm compliance with GDC 61.

As noted above, GDC 61 also requires that the fuel storage system be designed with appropriate filtering capability. The PSAR indicates that the FPC is designed with the capability of removing radioactive materials, corrosion products, and impurities from the fuel pool water. The NRC staff finds that this identified design criterion conforms with the requirements of GDC 61 as it relates to filtering systems associated with fuel cooling and storage.

The NRC staff evaluation of the fuel pool design against the requirements of GDC 61 are discussed in Sections 9.1.52 and 9.2.5 of this report. The NRC staff evaluated the additional design features of the FPC identified above from PSAR Sections 3.1.6.2 and 9.1.3 and find that requiring the system to be capable of maintaining the bulk pool temperature below 140°F and indicating the minimum decay heat load removal capability is consistent with the requirements of GDC 61 as it related to providing adequate decay heat removal capability.

In addition to the design features evaluated in Section 9.1.2 of this report, PSAR Section 9.1.3.5 indicates that the FPC is designed with no drains, piping, or other systems that would allow fuel pool water to drain below the minimum level needed to support plant safety analyses. The PSAR also indicates that the refuel floor in the Refueling Platform area is monitored by a radiation monitor alarm which initiates closure of the RB isolation dampers and secures the RB upper-level supply air handling units (AHUs) in the event of a fuel handling accident.

The NRC staff finds that the FPC has included adequate design criteria that, if satisfied, would prevent drainage of the fuel pool and suitable shielding for the refueling floor area, which are consistent with the requirements of GDC 61 as it relates to suitable shielding for radiation protection and the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions. Therefore, staff finds that the applicant has identified adequate design criteria for the FPC that conform with the requirements of GDC 61.

<u>GDC 63—Monitoring fuel and waste storage, as it relates to detecting conditions that may result in loss of residual heat removal capability and excessive radiation levels.</u>

Under GDC 63, applicants shall include appropriate systems in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels, and (2) to initiate appropriate safety actions. In PSAR Section 9.1.3.5 the applicant indicates that the FPC is provided with monitoring equipment that provides alarms and indications locally and to a continuously manned location. These instruments monitor pool water level, pool temperature and local radiation levels. The equipment is frequently tested. Additional information about the radiation monitoring instrumentation is provided in PSAR Section 11.5 and is evaluated by the NRC staff in Section 11.5 of this report.

In PSAR Section 9.1.3.7, the applicant indicates that the fuel pool level instrument provides remote wide-range fuel pool level monitoring in accordance with RG 1.227 (NRC 2019-TN13278), Wide-Range Spent Fuel Pool Level Instrumentation.

The NRC staff evaluated the description of the FPC capability for monitoring pool conditions that may result in the loss of residual heat removal capability and excessive radiation levels and determined that the applicant has identified adequate design criteria that conforms with the requirements of GDC 63. RG 1.227 provides design criteria, the NRC staff considers acceptable for level instrumentation that could be credited with meeting monitoring requirements identified in 10 CFR 50.155 “Mitigation of beyond-design-basis events.” 10 CFR 50.155(e) requires each operating power reactor licensee to provide reliable means to remotely monitor wide-range water level for each SFP at its site until 5 years have elapsed since all of the fuel within that SFP was last used in a reactor vessel for power generation.

The NRC staff also determined that designing the FPC wide-range level instrumentation in conformance with RG 1.227 would allow the level instrumentation to be in conformance with 10 CFR 50.155.

10 CFR 20.1101, “Radiation Protection Programs”

Compliance with 10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.

In PSAR Section 9.1.2.3, the applicant indicates that it is following the guidance provided in RG 1.13 and ANSI/ANS-57.2 paragraph 5.1.5, for maintaining safe radiation levels for personnel during anticipated operating and accident conditions. Additional information on the design of SSCs relative to ALARA is presented in Section 12.3. The NRC staff evaluation of this information is discussed in Section 12.3 of this report.

9.1.3.4 Conclusion

The NRC staff evaluated the fuel pool system design description provided in PSAR 9.1.3 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that a fuel pool cooling system constructed in accordance with these criteria could meet the requirements of 10 CFR 20.1101(b), 10 CFR 50.155(e), and GDC 2, 4, 5, 61, and 63 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

The applicant will complete the thermal analysis of the spent nuclear fuel pool, to demonstrate that the facility meets the applicable requirements of GDC 2, 4, 5, 61, and 63 of Appendix A of 10 CFR Part 50, 10 CFR 50.68 or 10 CFR 70.24 (TN4883) and make it available for staff review and approval as part of, the OLA at the CRN Site. The relevant regulations require the applicant to submit the FSAR in support of an OLA, which should include the key operational variables (for example, pool thermal load, water level and temperature) that ensure that the pool conditions are bounded by the thermal analysis assumptions.

9.1.4 Light Load Handling System

9.1.4.1 Introduction

The light load handling system (LLHS) handles, moves, and stores fuel assemblies and control rod assemblies during fuel transfer operation. The LLHS system is an integrated system of equipment and tools for refueling, handling, and storing fuel assemblies from receipt of the new fuel shipping container to shipment of the spent nuclear fuel cask. The safety objective of the system is to avoid criticality accidents, releases of radioactivity as a result of damage to irradiated fuel, and unacceptable personnel radiation exposures.

9.1.4.2 Regulatory Evaluation

SRP Section 9.1.4, Revision 4, "Light Load Handling System and Refueling Cavity Design," issued July 2014, describes the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections:

- GDC 2, as it relates to SSCs important to safety being capable of withstanding the effects of earthquakes.
- GDC 5, as it requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units.
- GDC 61, as it relates to fuel storage and handling systems being designed to ensure adequate safety under normal and postulated accident conditions.
- GDC 62, as it relates to preventing criticality in the fuel storage and handling system, preferably by use of geometrically safe configurations.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.1.4.3 Technical Evaluation

The PSAR describes the LLHS as consisting of the SSCs credited for handling new and spent nuclear fuel during its lifecycle (receipt, refueling, and loading into fuel casks). The LLHS uses equipment during fuel movement and refueling operations designed to lift loads weighing up to and including a single fuel assembly and its handling device. The application indicates that the LLHS does not perform a Safety Category 1, 2, or 3 function.

The PSAR Section 9.1.4.4 identifies the major components that are part of the LLHS. The components include:

- Fuel prep machine, which includes two fuel prep machines for handling fuel assemblies while removing or installing channels on the fuel bundles.
- Channel handling tool, which is used in conjunction with the fuel preparation machine to remove, install, and handle fuel channels in the fuel pool.
- Channel transfer grapple, which is an air-actuated device consisting of a frame, air cylinder, and two jaws. It is used with the refueling platform auxiliary hoist to transport individual irradiated fuel channels between working and storage facilities in the fuel pool.
- Channel handling boom, which supports the channel handling tool and load balancer over the fuel prep machine during channel removal and installation operations.
- Refueling platform, consists of a bridge that spans the width of the fuel pool and reactor cavity, a trolley rides on the bridge. The refueling platform rides on two rails set into the refuel floor.

GDC 2, "Design Basis for Protection against Natural Phenomena"

The NRC staff reviewed the LLHS for compliance with the requirements of GDC 2, with respect to its design for protection against the effects of earthquakes. One method of complying with the requirements of GDC 2 is by following the guidance in RG 1.29 (NRC 2021-TN12804), rev 6, Regulatory Position C.1. This provision provides guidance on determining which SSCs shall be classified as seismic Category I.

The LLHS is located on top of the SFP, which is located inside the RB. The RB is a seismic Category I structure design to withstand the effect of natural phenomena and provide protection to all SSCs located inside it. Additionally, in PSAR Section 9.1.4.5 the applicant indicated that the refueling platform is designed in accordance with RG 1.29, rev 6., Positions C.1.d bullet 4, and C.2. The applicant stated that the refueling platform is designed such that no single component failure will result in the dropping or damaging of a fuel assembly on the fuel hoist. In PSAR Appendix 3a "Preliminary Classification of Structures, Systems, and Components," Table 3A-1, "Preliminary BWRX-300 Component Classification List," the applicant indicates that the majority of the LLHS is designed as non-seismic components. The table indicates that the refueling platform seismic classification will be provided in the FSAR.

The NRC staff reviewed the LLHS design description provided in PSAR Section 9.1.4.5 and finds that designing the LLHS components, apart from the refueling platform, in a manner that no single component failure would cause a criticality accident or a release of radioactivity would be an adequate design. This would mean the LLHS is correctly categorized and consistent with the requirements of GDC 2. The refueling platform is a large component that travels above the fuel in the pool and the reactor vessel; therefore, the applicant has indicated that it is designed in accordance with RG 1.29, Positions C.1.d bullet 4, and C.2. The NRC staff finds that designing the refueling platform in accordance with RG 1.29, Positions C.1.d bullet 4, and C.2 is an acceptable design criterion to demonstrate compliance with the requirements of GDC 2. The NRC staff noted that the applicant has not identified the seismic design classification of the refueling platform in the PSAR. The applicant indicated that this classification will be provided in the FSAR. The applicant was not required to make this classification at the CP stage. At the FSAR review stage, should the applicant apply for an OL, the NRC staff will confirm that the final seismic design of the refueling platform is consistent with RG 1.29, Positions C.1.d bullet 4, and C.2, as indicated in the PSAR.

GDC 5—Sharing of structures, systems, and components

GDC 5 requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units.
The CRN-1 Site is a single unit, single module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the LLHS would be in conformance with GDC 5.

GDC 61—Fuel storage and handling and radioactivity control, as it relates to handling of radioactive materials

Under GDC 61, the fuel storage and handling system shall be designed to assure adequate safety under normal and postulated accident conditions. As described in Section 9.1.4 of the SRP, the NRC staff reviewed the LLHS system for compliance with the requirements of GDC 61 with respect to its design for protection against releases of radioactivity to the environment as a result of fuel damage and avoidance of excessive personnel radiation exposure. As described in Section 9.1.4 of the SRP, one method of complying with the requirements of GDC 61, in part, is by following the guidelines of ANSI/ANS 57.1-1992, as endorsed by RG 1.13, rev 2 (NRC 2007-TN13096).

In PSAR Section 9.1.4.5 the applicant indicates that the LLHS includes features to prevent release of radioactivity or excessive personnel radiation exposures. The refueling platform is provided with a position-indicating system and travel limit computer that prevent collision of the traveling fuel assembly with pool obstacles. The system also controls where a loaded fuel mast can travel, its loading, and speed for all areas of travel. The LLHS is also provided with travel interlocks to ensure adequate water covering is maintained in order to maintain suitable shielding from the assembly in transit.

Based on the above evaluation, the NRC staff finds that the LLHS system design includes design features in conformance with guidelines of ANSI/ANS 57.1-1992, to which the PSAR states the design conforms to. Therefore, the NRC staff has determined that the applicant has identified adequate design features such that the design would conform to the requirements of GDC 61 with respect to the prevention of radioactivity release as a result of fuel damage from mishandling or failure of the LLHS system.

GDC 62, “Prevention of Criticality in Fuel Storage and Handling”

GDC 62 requires that criticality in the fuel storage and handling system be prevented by physical systems or processes, preferably by use of geometrically safe configurations. The NRC staff reviewed the LLHS system for compliance with the requirements of GDC 62, with respect to the prevention of criticality in fuel handling systems. As discussed in Section 9.1.4 of the SRP, one method of complying with the requirements of GDC 62, in part, is following the guidelines of ANSI/ANS 57.1-1992.

In PSAR Section 3.1.6.3, the applicant indicates that the LLHS includes interlocks that detect conditions of the refueling equipment and the control rods. These interlocks are credited to enforce the operational procedures for fuel handling that prevent criticality by restricting movement of the refueling platform and control rods. The applicant stated that the fuel handling equipment is designed to prevent inadvertent criticality and to maintain shielding and cooling of spent nuclear fuel as necessary to meet operating and offsite dose constraints. In PSAR Section 9.1.A.3 “Safety Evaluation,” the applicant indicates that the LLHS conforms to the applicable guidance of ANSI/ANS-57.1.

The NRC staff evaluated the design criteria identified above and finds that the design criteria and interlocks identified are consistent with the guidance provided in SRP 9.1.4, and the guidelines of ANSI/ANS 57.1-1992.

Based on the above evaluation, the NRC staff finds that the LLHS system design includes design features consistent with guidelines of ANSI/ANS 57.1-1992. Therefore, the NRC staff has determined that the applicant has identified adequate design features in conformance with the requirements of GDC 62 with respect to the prevention of criticality.

9.1.4.4 Conclusion

The NRC staff evaluated the LLHS design description provided in PSAR Section 9.1.4 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that an LLHS constructed in accordance with these criteria could meet the requirements of GDC 2, 5, 61, and 62 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

The applicant will complete the seismic design of the refueling platform to demonstrate that the facility meets the applicable requirements of GDC 2 and make it available for staff review and approval as part of, the OLA at the CRN Site, should the applicant apply for an OL. The regulations require the applicant to identify in the FSAR the seismic design of the refueling platform.

9.1.5 Overhead Heavy Load Handling System

9.1.5.1 Introduction

The overhead heavy load handling system (OHLHS) consists of all equipment for moving all heavy loads (i.e., loads weighing more than one fuel assembly and control rod assembly) at the plant site. The review focuses on critical load handling, during which inadvertent operations or equipment malfunctions, separately or in combination, could cause a release of radioactivity, criticality accident, or inability to cool the fuel within the reactor vessel or SFP, or could prevent safe shutdown of the reactor.

The application indicates that the OHLHS does not perform a Safety Category 1, 2, or 3 function. The PSAR only includes the design criteria for one OHLHS system, the RB polar crane. PSAR Section 9.5.9 indicates that design criteria for additional OHLHS will be provided in the FSAR. At the FSAR review phase, the NRC staff will evaluate the additional OHLHS against the design requirements discussed in this section. This is acceptable because these cranes are not credited to perform a safety-related function or credited for the mitigation of a postulated accident scenario, additionally, the PSAR indicates that any additional OHLHS will be design in accordance with the design criteria identified in Section 9.1.5. Therefore, the applicable regulations do not require the specific design criteria for the additional OHLHS to be submitted at the CP stage.

9.1.5.2 Regulatory Evaluation

SRP Section 9.1.5, Revision 1, "Overhead Heavy Load Handling Systems," issued March 2007, gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections:

- GDC 1, "Quality Standards and Records," as it relates to the design, fabrication, and testing of SSCs important to safety to maintain quality standards.
- GDC 2, as it relates to the ability of structures, equipment, and mechanisms to withstand the effects of earthquakes.
- GDC 4, as it relates to the protection of safety-related equipment from the effects of internally generated missiles (i.e., dropped loads).
- GDC 5, as it relates to the sharing of equipment and components important to safety among multiple operating units at one single site.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.1.5.3 Technical Evaluation

The OHLHS consists of the components and equipment necessary for the safe handling of heavy loads such as the vessel head or a spent nuclear fuel cask. SRP Section 9.1.5 defines heavy loads as loads weighing more than the weight of one fuel assembly plus its handling device. The PSAR defined a heavy load in the same manner but did not indicate the actual weight limit that this represents. The applicant was not required to submit this information at the CP phase. The NRC staff reviewing the FSAR will confirm that the actual weight limit conforms with this definition. The NRC staff finds the proposed definition is consistent with SRP Section 9.1.5.

In PSAR Section 9.1.5.2 the applicant indicates that the proposed design follows the guidance of SRP 9.1.5 for overhead heavy load handling systems and compliance with GDC 1, 2, 4 and 5.

GDC 1, "Quality Standards and Records"

The NRC staff reviewed the OHLHS for compliance with GDC 1, which requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. As discussed in Section 9.1.5 of the SRP, one method of complying with the requirements of GDC 1, in part, is consistency with the guidelines of NUREG-0554 (NRC 1979-TN13281), as supplemented by the criteria of American Society of Mechanical Engineers (ASME) NOG-1-2004 for Type 1 cranes. NRC documented their endorsement of ASME NOG-1-2004 in "NRC Regulatory Issue Summary 2005-25, Supplement 1 Clarification of NRC Guidelines For Control of Heavy Loads." The guidelines of NUREG-0554 and the criteria of ASME NOG-1 for Type 1 cranes include provisions for design, installation, inspection, testing, and maintenance of cranes. As discussed in SRP Section 9.1.5, consistency with the criteria in ASME NOG-1 is an acceptable method of demonstrating consistency with NUREG-0554. In RG 1.244 (NRC 2021-TN13113) the NRC staff endorsed ASME NOG-1-2020, the NRC endorsement of ASME NOG-1–2020 updates the guidance in NUREG-0554.

The PSAR indicates that the RB polar crane and its associated main and auxiliary hoist are designed as single-failure-proof components in accordance with the guidelines of ASME NOG-1-2020 criteria for a Type 1 crane.

Therefore, based on that consistency with acceptable guidance, the NRC staff finds the applicant has identified adequate design criteria for the OHLHS consistent with the requirements of GDC 1.

GDC 2, "Design Basis for Protection against Natural Phenomena"

The NRC staff reviewed the OHLHS for compliance with the requirements of GDC 2, with respect to its design for protection against the effects of earthquakes. As discussed in SRP Section 9.1.5, one method of complying with the requirements of GDC 2 is consistency to Regulatory Position C.2 of RG 1.29 and the guidelines in Section 2.5 of NUREG-0554. In RG 1.244, Revision 0, the NRC staff endorsed ASME Standard NOG-1–2020 as adequate design criteria for cranes used as part of a highly reliable handling system. In RG 1.244, the NRC staff determined that NOG-1 Type I cranes provide a more comprehensive standard for the design, fabrication, and preoperational testing of these cranes that better incorporates knowledge from crane manufacturers, crane users, and the NRC staff, among other stakeholders, than the standard included in NUREG-0554.

In PSAR Section 9.1.5.3 the applicant indicated that the RB polar crane is designed according to ASME NOG-1-2020 for a Type I single-failure proof crane. Such cranes include redundancy and other features to ensure that the failure of a single component in the load path does not result in the loss of capability to stop and hold the critical load.

The PSAR indicates that the polar crane is designed as a single-failure-proof crane, with mechanical stops, electrical interlocks, well-defined safe load paths, established load handling procedures, and a plant configuration that provides redundancy and duality in certain components, so the probability of load drops is minimized. The polar crane is designed to retain its load throughout an SSE. The polar crane is not credited for mitigation of accident events and is therefore designed as a Seismic Category II structure in accordance with RG 1.29, Position C.2. The PSAR indicates that the polar crane complies with ASME NOG-1-2020 paragraph 6170 and Table 7210-1 which specifies inspection, testing, and enhanced quality assurance requirements for various electrical, mechanical, and structural subcomponents.

The NRC staff has reviewed the design criteria for the polar crane presented in PSAR Section 9.1.5 and determined that the applicant has identified the adequate design criteria based on previously endorsed industry standards that conform to the requirements of GDC 2.

GDC 4, "Environmental and Dynamic Effects Design Bases"

The NRC staff reviewed the OHLHS for compliance with the requirements of GDC 4 with respect to protection of fuel and safety-related equipment from the effects of internally generated missiles (dropped loads). A dropped heavy load in a critical area could cause a release of radioactive materials, a criticality accident, or inability to cool fuel within the reactor vessel or SFP or could prevent safe shutdown of the reactor. RG 1.13, Rev.2 (NRC 2007-TN13096), Regulatory Position C.5, provides the following guidance for meeting these requirements in spent nuclear fuel storage areas:

> Cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving over the pool. Furthermore, the spent nuclear fuel storage facility design should have at least one of the following provisions with respect to the handling of heavy loads, including the spent nuclear fuel cask:
>
> a) Cranes should be designed to provide single-failure-proof handling of heavy loads, so that a single failure will not result in the crane handling system losing the capability to perform its safety function.
> b) The spent nuclear fuel cask-loading area should be designed to withstand, without significant leakage of the adjacent spent nuclear fuel storage, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted.

The PSAR indicates that the polar crane has been designed as a single-failure-proof crane, in conformance with the criteria of ASME NOG-1-2020 for Type 1 cranes. SRP 9.1.5.III indicates that the general objective of the review of the OHLHS is to confirm that either the OHLHS has been designed for a highly reliable handling system or the potential consequences of a dropped load have been evaluated for acceptable consequences. The NRC staff has previously determined that an OHLHS that meets the criteria of ASME NOG-1-2020 for Type 1 cranes is, therefore, considered a highly reliable handling. Therefore, the NRC staff finds that the applicant is not required to evaluate the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted.

In addition, one method of reducing the probability and mitigating the consequences of an accidental load drop, as discussed in SRP Section 9.1.5.III.3, is a description of a heavy load handling program that is consistent with the guidelines on general programmatic controls for the design, operation, testing, maintenance, and inspection of heavy load handling systems, with a particular emphasis on establishing safe load paths for critical load handling. In SRP 9.1.5.I.3, the NRC staff identified that acceptable general programmatic guidelines for design, operation, testing, maintenance, and inspection are those in conformance with Section 5.1.1 of NUREG.0612 (NRC 1980-TN9395).

The PSAR states that:

> The guidelines of ASME NML-1 are applied to address the program for conducting normal handling system operations, including maintenance, surveillance, and periodic testing of handling system SSCs. The heavy load lift program for normal operations includes development of lift procedures, crane operator and rigger qualifications, crane inspection, testing, and maintenance, and selection and use of lifting devices not specially designed. The guidelines of ASME NML-1 are also applied to establish criteria for protecting safety functions during nuclear safety critical lifts. These criteria include control of load motion by design and interlocks, enhanced safety handling systems, and engineering controls that include analysis of a postulated load drop. Additionally, the PSAR specifies that the polar crane is also designed with a system of interlocks that prevent movement outside safe load paths.

In RG 1.244, “Control Of Heavy Loads At Nuclear Facilities”, Revision 0, the NRC staff endorsed with clarifications ASME Std. NML-1–2019, which provides programmatic guidance that specify information needed to demonstrate that safety functions would be appropriately protected against handling system equipment failures. The NRC endorsement of ASME Std. NML-1–2019 in this RG updates the guidance in NUREG-0612. As the applicant has committed to following ASME NML-1-2019, the NRC staff finds that the applicant has identified adequate design criteria that conform with the requirements of GDC 4.

GDC 5 – Sharing of structures, systems, and components

GDC 5 requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units.

The CRN Site is a single unit, single module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the OHLHS design can be in conformance with GDC 5.

9.1.5.4 Conclusion

The NRC staff evaluated the OHLHS design description provided in PSAR Section 9.1.5 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that an OHLHS constructed in accordance with these criteria could meet the requirements of GDC 1, 2, 4, and 5 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.1.A Criticality Safety of New and Spent Fuel Storage

9.1.A.1 Introduction

PSAR Section 9.1.A describes the standards, design criteria, and conceptual design of new and spent nuclear fuel storage that provide assurance that, once the new and spent nuclear fuel pool rack design is finalized, stored fuel will remain subcritical consistent with requirements of GDC 62 and 10 CFR 50.68 (TN249).

9.1.A.2 Regulatory Evaluation

The NRC staff finds the following regulations applicable to this section:

- 10 CFR 50.34, “Contents of applications; technical information,” including:
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include “Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.”
- 10 CFR 50.35.

10 CFR 50.68(b) provides requirements related to the prevention of criticality accidents in lieu of requirements to maintain a monitoring system capable of detecting criticality as described in 10 CFR 70.24 (TN4883).

GDC 62 requires that criticality in fuel storage and handling systems be prevented by physical systems or processes, preferably by the use of geometrically safe configurations.

PSAR Section 9.1.A.1 states, in part, that “in lieu of a monitoring system capable of detecting a criticality, the fuel storage facilities comply with the additional design and analysis requirements specified in 10 CFR 50.68(b).” The regulation at 10 CFR 50.68(b) provides an alternative to 10 CFR 70.24, which requires a monitoring system to provide an audible alarm when an accidental criticality occurs in an area containing specific quantities of special nuclear material. Section 50.68 applies to holders of CPs and 10 CFR 50.68(b) provides alternative requirements that a licensee can implement in lieu of installation of a criticality monitoring system that would otherwise be required by 10 CFR 70.24. In its PSAR, TVA states that it will elect to comply with the requirements in 10 CFR 50.68(b) rather than the requirements of 10 CFR 70.24.

The requirements of 10 CFR 70.24 are only applicable to licensees authorized to possess special nuclear material (SNM). The NRC staff did not evaluate whether requirements in 10 CFR 50.68, “Criticality accident requirements,” or 10 CFR 70.24, “Criticality accident requirements,” would be met for the construction of CRN-1 because TVA is not requesting to possess SNM, nor would the CP authorize TVA to possess SNM. Instead, the NRC staff assessed whether TVA identified relevant requirements and descriptions of the preliminary facility design, to determine whether the PSAR provides an acceptable basis for the development of systems and whether there is reasonable assurance that TVA will comply with 10 CFR 50.68 once CRN-1 possesses the relevant material. The NRC staff will need to evaluate whether or not the final fuel storage rack design and criticality analysis meet the requirements of 10 CFR 50.68 prior to authorizing TVA to possess SNM.

9.1.A.3 Technical Evaluation

PSAR Section 9.1.A.1 states that the design of new and spent nuclear fuel storage will apply, as applicable, RG 1.13, Revision 2 (NRC 2007-TN13096), “Spent Fuel Storage Facilities at Nuclear Power Plants,” and RG 1.240, Revision 0 (NRC 2021-TN13279), “Fresh and Spent Fuel Criticality Analysis.” RG 1.13, Revision 2, and RG 1.240, Revision 0, endorse the 1983 revision of ANSI/ANS-57.2, and Revision 4 of NEI 12-16, respectively, with additions, clarifications, and exceptions. NEI 12-16, Revision 4, with clarifications and exceptions discussed in RG 1.240, describes criticality analysis which can demonstrate compliance with requirements in 10 CFR 50.68(b). As noted in RG 1.240, Revision 0, RG 1.13, Revision 2 is relevant in that it discusses SSCs that are relied on by the nuclear criticality safety analysis. PSAR Section 9.1.A.1 also stated that design of new and spent nuclear fuel storage will apply ANSI/ANS 57.1, as applicable. SRP Section 9.1.1, “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” notes that ANSI/ANS 57.1 provides guidance acceptable to staff for meeting requirements associated with fresh and spent nuclear fuel storage and handling.

PSAR Section 9.1.A.3 states that the proposed design and criticality analysis of fuel storage racks will be consistent with that described in Section 3.5 of NEDE-24011-P-A-31, Revision 31, “General Electric Standard Application for Reactor Fuel (GESTAR II).” Section 3.5 of NEDE-24011-P-A-31 provides requirements on in-core lattice k_∞ and fuel loading for BWR/2 through BWR/6 designs to ensure that regulatory requirements for fresh and spent nuclear fuel criticality are met. Requirements are provided for specific rack designs, and k_∞ values are calculated in the normal reactor core configuration. NRC staff previously reviewed this methodology using the guidance in DSS-ISG-2010-01, “Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools,” which was the predecessor to RG 1.240. The NRC staff’s safety evaluation of these requirements for BWR/2 through BWR/6 fuel storage racks and core configurations are in ML17066A346. Analysis supporting requirements for BWRX-300 fresh and spent nuclear fuel criticality will be reviewed by the NRC staff after the fuel storage rack design is finalized, and prior to authorizing TVA to possess SNM, as described below.

Staff considered the applicability of methods used to develop NEDE-24011-P-A-31 requirements to the BWRX-300 design. The CRN-1 equilibrium cycle uses GNF2 fuel, and the NRC staff has accepted application of spent fuel pool criticality criteria in NEDE-24011-P-A-31 to GNF-2 fuel in BWR/2 through BWR/6 plants. CRN-1 enrichment, loading of gadolinium, and configuration of part-length rods are within the range of previously analyzed fuel.

Reactor operating parameters can also affect spent fuel composition, which is analyzed in the development of NEDE-24011-P-A-31 requirements. The NRC staff expects that CRN-1 moderator temperature, fuel temperature, power density, and control rod usage will be within the ranges that have been modeled previously. CRN-1 is designed with a higher nominal core exit void fraction than BWR/2 through BWR/6 designs, and natural recirculation may contribute to uncertainty in the prediction of void fraction. However, lattice depletion calculations at a range of void fractions are part of the methodology used to develop NEDE-24011-P-A requirements, and the applicant has committed to follow RG 1.240, which includes guidance that the full range of void fractions should be considered in conjunction with other parameters. Therefore, application to CRN-1 may include depletion analysis with the existing methodology at higher void fraction. Non-baseload operation may also alter reactor operating parameters, depending on the frequency and magnitude of core power changes, and require additional analysis or justification.

Spent fuel pool criticality analysis also includes consideration of normal, abnormal, and accident conditions in the fuel rack. Applicable conditions depend on the fuel rack design. While these calculations must be reviewed when the applicant applies for a license to possess nuclear fuel, and similarity of the finalized fuel rack design to previous designs may affect the required level of review, staff is not aware of issues requiring incorporation of novel features into the CRN-1 fuel racks.

Based on the above, staff finds there is reasonable assurance that the applicant will be able to resolve questions concerning the applicability of NEDE-24011-P-A-31 criticality analysis methodologies to CRN-1 prior to completion of construction. PSAR Section 9.1.A.3.1 describes the analysis code that will be used to evaluate in-core lattice k_∞ and its validation. It states that racks will be analyzed using Monte Carlo N-Particle transport program (MCNP) and ENDF/B-VII.0 cross-section libraries. MCNP is a general use code for analysis of particle transport that is maintained by Los Alamos National Laboratory. Criteria in Section 3.5 of NEDE-24011-P-A-31 were developed using a Global Nuclear Fuel (GNF) proprietary version of MCNP as discussed in ML13122A423. Analysis methodology and validation, including the specific MCNP version and proprietary modifications, will be reviewed by staff once the finalized criticality analysis is provided.

As stated above, when the application requesting authorization to possess SNM at the CRN Site is submitted, staff will evaluate the CRN-1 fuel storage rack design and criticality analysis to ensure it is completed according to the regulatory requirements such that there is reasonable assurance that stored fuel will remain subcritical, and to ensure that any necessary restrictions, such as restrictions on the configuration of stored fuel, are incorporated into the CRN-1 licensing basis.

9.1.A.4 Conclusion

The NRC staff recognizes that Section 9.1.A is not complete for the NRC staff to make 10 CFR 50.68 and GDC 62 safety findings. The applicant has identified appropriate regulatory requirements and guidance and indicates that they will perform analysis consistent with that performed to support BWR/2 through BWR/6 designs, which demonstrates a framework for the future design of the new and spent nuclear fuel storage racks. The NRC staff finds this level of preliminary information sufficient to support issuance of the CP pursuant to the regulations of 10 CFR 50.34(a)(3) 10 CFR 50.35, as applicable, given that TVA will not be authorized to possess SNM at the CRN Site under the CP. TVA will be required to demonstrate compliance with either 10 CFR 70.24 or 10 CFR 50.68 to obtain authorization to possess relevant quantities of SNM at the CRN Site.

9.2 Water Systems

9.2.1 Service Water Subsystem

9.2.1.1 Introduction

This section discusses the service water (SW) subsystem. SW provides cooling water to the plant cooling water (PCW) system (Section 9.2.2) heat exchangers and is a subsystem of the circulating water system (CWS) (Section 10.4.5). The heat from the SW subsystem is transferred to the environment using normal heat sink (NHS) (Section 9.2.9) cooling tower with provisions to directly use tie-in to NHS piping to provide once-though cooling when the cooling tower is unavailable.

9.2.1.2 Regulatory Evaluation

The NRC staff reviewed the CRN-1 proposed design for service water system in accordance with SRP Section 9.2.1, Revision 5. The SRP identifies the following as the relevant regulatory requirements:

- GDC 2, as it relates to the capability of the design to maintain and perform its safety function following an earthquake.
- GDC 4, as it relates effects of missiles inside and outside of containment, pipe whip, jets, and environmental conditions from high- and moderate-energy line brakes and dynamic effects of flow instabilities and loads (e.g., water hammer) during normal plant operation as well as during upset or accident conditions.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink.
- GDC 45, “Inspection of cooling water system,” as it relates to the design provisions to permit appropriate periodic inspection of important components and equipment.
- GDC 46, “Testing of cooling water system,” as it relates to the design provisions to permit appropriate periodic pressure and functional testing of components and equipment.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, “Contents of applications; technical information,” including (TN249):

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

- – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include “Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.”
- 10 CFR 50.35.

9.2.1.3 Technical Evaluation

The SW subsystem is a cooling system described in PSAR Section 9.2.1 and shown in Figure 9.2-1 consisting of two 100 percent capacity pumps to provide cooling water to the PCW heat exchangers for normal operating conditions. The SW pump normal source of inventory is from the cooling tower basin with previsions to tie-in SW piping to take inventory from a once-through cooling source when NHS cooling tower is unavailable. Two individual SW subsystem pump discharge lines each contain pressure instrumentation, air release valves, and an isolation valve. Each SW subsystem line has a strainer at the pump discharge upstream of the PCW heat exchangers. According to the applicant, the PCW heat exchangers are designed to reduce the possibility of leakage from PCW to the SW subsystem. If leakage were to occur, the SW subsystem is monitored for radioactivity and can be isolated if high activity level is detected. Screens are included between the cooling tower basin and the SW subsystem pump suction lines. The SW subsystem and its piping are located in Turbine Building (TB) and outside yard.

Section 9.2.1.1 of PSAR defines the SW subsystem functions to reject heat from the PCW heat exchangers to the environment through the NHS system for Safety Class 3 (SC3) and non-safety- equipment. As indicated in PSAR Section 9.2.1.3, the SW subsystem is not credited for mitigation of DBA and has no safe shutdown functions. During normal operation, only one PCW heat exchanger is typically in service, and the corresponding SW subsystem pump is used to provide cooling water flow. During a plant shutdown, both PCW heat exchangers and SW subsystem pumps are typically in service to facilitate a plant shutdown. The SW subsystem also supports PCW cooling function in event of loss-of-preferred power (LOPP), if the non-safety standby diesel generator (SDG) is available.

PSAR Table 3A-1 defines classification of the service water system as SC3 and non-safety classification (SCN) for pumps, heat exchangers, and piping associated with heat removal and system function.

The NRC staff evaluated the sufficiency of the CRN-1 preliminary design features for systems and components, as described in PSAR Section 9.2.1, for the issuance of a CP using Section 9.2.1, "Station Service Water System," of NUREG-0800 (NRC 2021-TN8013).

The NRC staff evaluated the applicant's compliance with the GDCs listed above. The applicant indicated in PSAR Section 9.2.1 that SW subsystem is not credited for mitigation of DBA and has no safe shutdown function. The LOPP event is presented in PSAR Section 15.5.4.2.2 indicating sequence of events and conclusion doesn't indicate use of SW subsystem. The applicant identified criteria for this non-safety system, based on the design and layout of the SW subsystem, to ensure that a failure of the subsystem will not adversely affect the functional performance of SC1 systems or components to meet GDC 2 requirements. Therefore, the NRC staff finds that the preliminary SW subsystem design is consistent with GDC 2 to withstand the effects of natural phenomena and support issuance of the CP.

The applicant intends to meet GDC 4 requirements by having none of the equipment in the SW subsystem, or cooled by the subsystem, being required to mitigate DBAs and by piping being evaluated in Section 3.6 of the PSAR to ensure failure of non-safety system will not impact function of safety-related SSCs. Since SW is not credited for DBAs and potential failure of piping is evaluated in Section 3.6, the NRC staff finds the preliminary SW subsystem is consistent with GDC 4 to accommodate the effects of environmental and dynamic effects during operation.

GDC 44, 45, and 46 require the cooling capability, inspection and testing of cooling water system. PSAR Section 9.2.1.3 indicates that the preliminary SW subsystem is not required to provide cooling to any SC1 components and is not credited as UHS in Chapter 15 safety analysis. The SW subsystem is designed to permit periodic inspection and testing to ensure the integrity and capability of the system to provide cooling to SSCs and to adequately transfer heat to the NHS. Based on this capability, the NRC staff finds that the SW subsystem is consistent with GDC 44, 45 and 46.

Although these GDC are indicated as not applicable in the PSAR, the applicant has indicated that the SW subsystem does transfer heat from the PCW to the NHS system under normal operating conditions, including an LOPP if equipment is available, and contains redundancy of trains to perform function, assuming single failure. At this stage of CP review the NRC staff is unable to make final determination that GDC 44, 45 and 46 apply, based on SW system potential credit or impact on accident analyses. The NRC staff discussed SW support of any Chapter 15 transient during audit and applicant indicated that SW is not credited to perform any function during PSAR Chapter 15.2.5 transients. If SW subsystem is later credited in Chapter 15 safety analysis, then additional review will be required during OL to ensure appropriate classification and applicability of GDC's to SW.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the service water subsystem design can be in conformance with GDC 5.

The applicant indicated in Table 3.1-1 of PSAR that SW subsystem complies with GDC 64 related to control of releases of radioactive materials to the environment. SRP 9.2.1 indicates that there should be provisions to detect and control leakage of radioactive contamination into and out of the system. The system is once-through system cooling a PCW which carries normally nonradioactive cooling water. The SW service water has radiation monitors installed on SW subsystem piping downstream of the PCW heat exchanger to detect leakage from the PCW to the SW subsystem. Based on the applicant's description of the SW system, the NRC staff finds the use of monitors provides reasonable control in SW system against radioactive release to the environment.

9.2.1.4 Conclusion

The NRC staff evaluated the service water subsystem design description in PSAR Section 9.2.1 and concluded that the applicant has proposed adequate design criteria for the system to support CP review. The NRC staff has confidence that a Service Water Subsystem system constructed in accordance with these criteria could meet the requirements of GDC 2, 4, 5, 44, 45, and 46 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.2.2 Plant Cooling Water System

9.2.2.1 Introduction

This section discusses the PCW. The PCW provides cooling water to SCN and SC3 components and provides a barrier against radioactive contamination of the SW subsystem (PSAR Section 9.2.1) and NHS system (PSAR Section 9.2.9). The PCW has two closed-loop piping subsystems, the reactor component cooling water piping distribution, and the turbine component cooling water piping distribution, which provide cooling water to various heat exchangers.

9.2.2.2 Regulatory Evaluation

The NRC staff reviewed the CRN-1 proposed design for PCW system in accordance with SRP Section 9.2.2 “Reactor Auxiliary Cooling Water System” (Revision 4), because this system performs function of providing reactor component cooling water. As identified in SRP Section 9.2.2, the relevant regulatory requirements are:

- GDC 2, as it relates to the capability of the system to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods.
- GDC 4, as it relates to effects of missiles inside and outside of containment, effects of pipe whip, jets, environmental conditions from high- and moderate-energy line breaks, and dynamic effects of flow instabilities and attendant loads (i.e., water hammer) during normal plant operation as well as upset or accident conditions.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink
- GDC 45, “Inspection of cooling water system,” as it relates to the design provisions to permit appropriate periodic inspection of important components and equipment.
- GDC 46, “Testing of cooling water system,” as it relates to the design provisions to permit appropriate periodic pressure and functional testing of components and equipment.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, “Contents of applications; technical information,” including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include “Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.”
- 10 CFR 50.35.

9.2.2.3 Technical Evaluation

Section 9.2.2 of the PSAR provides a summary description of the PCW. The PCW consists of two piping subsystems; reactor component cooling water piping distribution and turbine component cooling water piping distribution which provide cooling water to various heat exchangers. The reactor component cooling water piping distribution consists of two redundant piping distributions that support RB equipment cooling functions, plant pneumatics system (PPS), and equipment associated with reactor cooling functions. The turbine component cooling water piping distribution is a single piping distribution that serves equipment in the TB and does not support reactor cooling functions. The following are defined in the PSAR as the Safety Category 3 functions:

- Provides cooling water to the FPC heat exchangers.

- Provides cooling water to isolation condenser pools cooling and cleanup system (ICC) heat exchangers.
- Provides cooling water for the shutdown cooling system heat exchangers.
- Provides cooling water for plant pneumatics system (PPS) compressor coolers.
- Provides a barrier against radioactive contamination of the SW subsystem and NHS system.
- Provides cooling water to SC3 TB components and coolers.

Table 3A-1 of the PSAR lists the preliminary classification of the PCW system as SC3 and SCN for pumps, heat exchangers, and piping associated with heat removal and system function.

The PCW contains two trains with each train consisting of a 100 percent capacity pump, a 100 percent capacity heat exchanger, one surge tank, connecting piping, control valves, and instrumentation. One train of PCW is normally in operation while the other is in standby. The heat from PCW is rejected to the NHS system by use of SW subsystem that interfaces with PCW through the PCW heat exchangers. The PCW is backed up by SDGs to support the system during LOPP, if available. Manual crosstie provisions placed upstream and downstream of each pump are available to provide operational flexibility.

The NRC staff evaluated the applicant’s compliance with the GDCs listed above. The applicant indicates in PSAR Section 9.2.2 that PCW and supported systems cooled by the PCW are not required to mitigate DBA and have no safe shutdown function. As indicated in PSAR Section 9.2.2.3, the design includes protection to ensure that a failure of the PCW system or components will not adversely affect nearby SC1 systems or equipment. Therefore, the NRC staff finds that the preliminary PCW system design is consistent with GDC 2 to withstand the effects of natural phenomena and support issuance of the CP.

The PCW system is a moderate energy piping system evaluated in PSAR Section 3.6.1 and designed to not adversely affect the function of SC1 components. PSAR Section 9.2.3 clarifies that PCW equipment is designed for the plant normal environmental conditions specified for its location (TB and RB) and not required to mitigate DBAs. Since PCW is not credited for DBAs and potential failure of piping is evaluated in Section 3.6, the NRC staff finds the preliminary PCW design presented in the PSAR is consistent with the requirements of GDC 4 and support of the CP.

The Clinch River Nuclear (CRN-1) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the service water subsystem design can be in conformance with GDC 5.

GDC 44, 45, and 46 require the cooling capability, inspection and testing of cooling water system. Section 9.2.2.3 of PSAR indicates that the PCW is designed to remove heat, as well as permit periodic inspection and testing, to ensure the integrity and capability of the system to cool SSCs and to adequately transfer heat from the systems indicated above under normal operating and normal cooldown conditions. Based on this capability, the NRC staff finds the PCW is consistent with GDC 44, 45, and 46.

Although these GDC are indicated as not applicable in the PSAR, the applicant has indicated that the PCW system does transfer heat from the PCW to the NHS system under normal operating conditions, including an LOPP if equipment is available, and contains redundancy of trains to perform function, assuming single failure. At this stage of CP review, the NRC staff is unable to make final determination that GDC 44, 45 and 46 apply, based on PCW system potential credit or impact on accident analyses. If PCW system is later credited in Chapter 15 safety analysis, then additional review will be required during OL to ensure appropriate classification and applicability of GDC's to PCW system.

The applicant further indicated that the PCW will interface with water systems that may contain radioactive material. PSAR Section 9.2.2.2 indicates, "Some systems cooled by the PCW contain fluid that has the potential to contaminate the PCW with radioactivity." To address potential radiation concerns, the PSAR clarifies that the PCW is designed as a closed-loop system to act as a buffer between potentially radioactive systems and the nonradioactive SW Subsystem that transfers heat to the environment. In addition, PSAR Section 9.2.1 indicates radiation monitoring of the SW subsystem is provided downstream of each PCW heat exchanger.

9.2.2.4 Conclusion

The NRC staff evaluated the PCW system in PSAR Section 9.2.2 and concluded that the applicant has proposed adequate design criteria for the system to support CP review. The NRC staff has confidence that a PCW system constructed in accordance with these criteria could meet the requirements of GDC 2, 4, 5, 44, 45, and 46, of appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.2.3 Demineralized Water Storage and Distribution Subsystem

9.2.3.1 Introduction

This section discusses demineralized water storage and distribution subsystem (DWS). The DWS is designed to treat raw water and distribute purified water to systems throughout the facility. The DWS is not safety-related, with the exception of containment isolation valve and corresponding piping. The DWS is not required for any design basis event (DBE) and has no safe shutdown function.

The DWS provides normal makeup for pool water evaporation in the ICC, boron injection system (BIS), chilled water equipment (CWE), liquid waste management system (LWM), and PCW system. The DWS does contain containment isolation valves, which are described in Section 6.2.4 of PSAR.

9.2.3.2 Regulatory Evaluation

The NRC staff determined that no current SRP section is directly applicable to the DWS (SRP Section 9.2.3, "Demineralized Water Makeup System," was withdrawn in December 1996). Consistent with the basis for withdrawing SRP Section 9.2.3, the NRC staff selected applicable portions of SRP Section 9.2.2, Revision 4, "Reactor Auxiliary Cooling Water System" and Section 9.2.6, Revision 3, "Condensate Storage Facilities," issued March 2007. The following regulatory requirements apply:

- GDC 2, as it relates to the capability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. .

  GDC 5, insofar as it requires that SSCs important to safety not be shared among power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions. The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, “Contents of applications; technical information,” including (TN249):

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include “Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.”

- 10 CFR 50.35.

9.2.3.3 Technical Evaluation

The DWS is comprised of the demineralized water storage tank, the demineralizer trailer which processes incoming raw water and provides makeup to the tank, redundant demineralized transfer pumps and associated piping, valves, and instrumentation. The distribution of the DWS includes providing water to systems in Control Building (CB), TB, Radioactive Waste Building (RWB), and RB.

The main function of the DWS is to provide purified water for normal makeup for pool water evaporation in the ICC, BIS, CWE, LWM, and PCW system. This system is normally operated during normal plant operation, and the system is not credited for mitigation of a DBA or has any safe shutdown functions. The NHS system provides the water source of demineralized water.

As indicted in Table 3A-1, “Preliminary BWRX-300 Component Classification List” (Sheet 11 of 22), the demineralized water piping and valves that form part of containment are classified as Safety Class 1, Quality Group B and Seismic Category I. All other components are classified as non-safety related.

The DWS provides makeup for systems within primary containment resulting in need for containment penetrations. As indicated in Table 6.2-2 “Process Line Isolation”, the DWS contains inside and outside containment isolation valves (CIV) to isolate process line, which are locked closed manual valves. Table 6.2-4 indicates these CIV are locked closed during normal, hot shutdown and post-accident conditions. In Section 6.2.4, the applicant provides information to address GDCs applicable to containment penetrations. The NRC staff analyzes this information in section 6.2.4 of this SE.

The NRC staff evaluated the applicant’s approach to comply with the GDCs listed above. GDC 2 establishes requirements with respect to SSCs important to safety for protection against the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. PSAR

9.2.3 does not show GDC 2 as applicable to the DWS. However, potential failure of non-safety SSCs should be evaluated for potential impact on components with safety-related functions. Therefore, the NRC staff will evaluate the final design's interfacing connections and piping and its applicability to GDC 2 at OL review stage. Leaving this for the OL phase is acceptable because, there complete design information is not required under NRC regulations to support issuance of a CP. The NRC staff finds that the PSAR has identified relevant requirements and provided sufficient description of preliminary design criteria of the DWS to support issuance of the CP.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the DWS design can be in conformance with GDC 5.

9.2.3.4 Conclusion

The NRC staff evaluated the DWS design description in PSAR Section 9.2.3 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that a DWS constructed in accordance with these criteria could meet the requirements of GDC 2 and 5 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

Since the non-safety-related DWS provides makeup water to the SC1 IC pools via the ICC system, the NRC staff will evaluate the final configuration as part of the OL review to confirm that the DWS interface with safety-related SSCs will not introduce any safety concerns resulting from a potential DWS failure or interface malfunction.

9.2.4 Potable Water Distribution and Sanitary Sewer Handling Subsystem

9.2.4.1 Introduction

This section discusses the potable water distribution and sanitary sewer handling subsystem. These systems are designed to provide potable water and handle sanitary sewage in areas outside of radiologically controlled areas. These subsystems are not safety-related and are not credited for the mitigation of postulated events and have no safe shutdown function.

The potable water distribution subsystem provides potable water for various SCN systems including the fire protection system (FPS) storage tanks, power block areas outside of radiologically controlled areas, and other users outside of the power block.

The sanitary sewer handling subsystem collects sanitary water from areas outside the radiologically controlled areas in the CB, the power block, and transfers it offsite.

9.2.4.2 Regulatory Evaluation

The NRC staff reviewed this system's design in accordance with the review procedures in SRP Section 9.2.4, "Potable and Sanitary Water Systems", Revision 3. The following regulatory requirement applies:

- GDC 60, as it relates to design provisions provided to control the release of liquid effluents containing radioactive material from contaminating potable and sanitary water systems.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.2.4.3 Technical Evaluation

The Potable Water Distribution Subsystem provides potable water to the plant and consists of supply line piping, valves, and instrumentation. The subsystem supplies potable water from the municipal potable water line and provides potable water to the FPS Fire Water Storage Tanks, power block areas outside of radiologically controlled areas, and other users outside of the power block (e.g., Administration Building). Potable water is supplied to the restroom facility within the MCR envelope, which is evaluated for flooding in the internal flooding analysis described in PSAR Section 3.4.1.

The Sanitary Sewer Handling Subsystem consists of piping and valves to collect sanitary water from the facility washrooms and break areas and transfers it off site to the municipality.

The NRC staff evaluated the applicant's compliance with the GDC listed above. As indicated in PSAR Section 9.2.4.3, the potable water distribution and sanitary sewer handling subsystems have no interconnection with radioactive water systems. Therefore, inadvertent contamination of potable and sewer water is minimized and GDC 60 is addressed for the proposed design.

9.2.4.4 Conclusion

The NRC staff evaluated the potable water distribution and sanitary sewer handling subsystem design description provided in PSAR 9.2.4 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that a potable water distribution and sanitary sewer handling subsystem constructed in accordance with these criteria could meet the requirements of GDC 60 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.2.5 Ultimate Heat Sink

9.2.5.1 Introduction

This section discusses the ultimate heat sink (UHS). The UHS consists of pools of water (shown in PSAR Figure 9.2-3) located within the RB to provide the Safety Category 1 function of heat sink for DBAs and off-normal conditions and used when the NHS system is not available. These pools include isolation condenser (IC) pools, the reactor cavity pool, the equipment pool, and the fuel pool.

The UHS performs various functions under normal and off-normal operating conditions. Cooling of the UHS pools is performed by the ICC system (described in Section 9.2.8) for the IC pools. The FPC system (Section 9.1.3) is used to maintain the Fuel pool, equipment pool, and reactor cavity pool inventory.

9.2.5.2 Regulatory Evaluation

The NRC staff reviewed the design of the UHS in accordance with SRP Section 9.2.5, “Ultimate Heat Sink” (Revision 3). The NRC staff’s acceptance of the design is based on compliance with the following requirements:

- GDC 2, as to capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink.
- GDC 45, “Inspection of cooling water system,” as it relates to the design provisions to permit inspection of components and equipment.
- GDC 46, “Testing of cooling water system,” as it relates to the design provisions to permit operational testing of components and equipment.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, “Contents of applications; technical information,” including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include “Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.”
- 10 CFR 50.35.

9.2.5.3 Technical Evaluation

The NRC staff evaluated the sufficiency of the CRN preliminary design features for systems and components, as described in PSAR Section 9.2.5, for the issuance of a CP using the applicable guidance and acceptance criteria from Section 9.2.5, “Ultimate Heat Sink,” of NUREG-0800. Staff acceptance of the UHS is based on meeting the requirements of GDC 2, 5, 44, 45, and 46.

The PSAR indicates the Safety Category 1 functions of UHS are:

- Water in IC pools provides heat removal from the Isolation Condenser System (ICS) under off-normal conditions.

- Water in the reactor cavity pool and the equipment pool provide heat removal from the passive containment cooling system (PCCS) under off-normal conditions.
- Water in the Fuel Pool provides heat removal from the fuel in the pool.
- The IC pools, reactor cavity pool, equipment pool, and fuel pool are part of the RB and are Seismic Category I structures to support the SC1 UHS.
- Other Safety Category 1 functions related to the ICC are described in Subsection 9.2.8 of the PSAR.

The PSAR also references the Safety Category 3 function of the UHS where the FPC provides heat removal and makeup capacity for the fuel pool, reactor cavity pool, and equipment pool to ensure the proper volume and temperature of water are available for the UHS function for these pools.

The UHS consists of a set of safety related pools (PSAR Figure 9.2-3) that contain the combined volume in the IC pools, fuel pool, reactor cavity pool, and equipment pool. The UHS is an integral part of the RB, and the pools are supported by the internal RB structure. The IC pool volumes are independent from the equipment pool, fuel pool, and reactor cavity pool. During normal operation, the NHS system transfers heat from plant equipment to the environment and IC pools are not used but remain available for removing heat from the reactor in event of accident through passive use of the isolation condensers within the pools.

The IC pools contain inventory for heat removal from the ICS during off-normal conditions. The component requirements for the IC inner and outer pools are defined in PSAR Section 9.2.8.2 and listed below:

- The smaller IC inner pool (IC Inner Pool A) and the full volume of the outer pools shall be capable of cooling the reactor for 72 hours after shutdown without replenishment.
- The larger IC inner pool (IC Inner Pool B + C) and the full volume of the outer pools shall be capable of cooling the reactor for 7 days after shutdown without replenishment.

Levels and temperatures in the IC pools, equipment pool, reactor cavity pool, and fuel pool will be monitored and maintained within administrative and technical specification limits, as indicated in Section 9.2.5.4 of PSAR. After 7 days, supplemental water can be provided through the IC pool diverse and flexible coping strategy (FLEX)/emergency mitigating equipment (EME) connection.

The reactor cavity pool and equipment pool inventory are used to support PCCS operation. The PCCS is a passive system for transferring heat by natural circulation from the containment atmosphere to these pools. When the plant is in normal operation, the equipment pool gate is removed, connecting the equipment pool and reactor cavity pool to provide the heat sink for the PCCS. The PCCS is described in PSAR Section 6.2.2 and transfers containment heat to the reactor cavity pool and equipment pool after a DBA. As stated in PSAR Section 9.2.5.3, the water volume in the equipment pool and reactor cavity pool is adequate to support PCCS for 7 days.

The UHS is an integral part of the RB structural boundary described in Section 3.8.4.1 of PSAR. General arrangement and approximate dimensions of the RB structures is illustrated in PSAR Figures 1.2-2, 1.2-3, 3.8-1, 3.8-8, and 9.2-3. The NRC staff is unable to locate any details or volumes to verify the 72 hours, and 7-day capacity of available inventory within the ICC pools for PCCS and IC performance. 10 CFR 50.34(a)(3)(iii) requires the PSAR submitted as part of a CPA to include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety." The PSAR has stated the preliminary volume is based on 72 hour and 7-day duration required to perform safety function. The NRC staff discussed specific volumes and physical dimensions during the audit which is based on numerous calculations. During the audit, the applicant summarized their calculations and volumes to support the capacities. The applicant also explained that the specific volumes and physical dimensions of the pools to support ICS and PCCS operation are not included in the PSAR UHS description because these values are expected to evolve during detailed design. The NRC staff finds the use of time duration to define pool inventory is adequate for CP review because the time duration is the basis for volume. It therefore provides enough information about approximate dimensions to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety and will review detailed information after it is finalized in the FSAR at the OL stage.

The fuel pool is defined as part of the UHS, which contains the spent fuel pool described in PSAR Section 9.1.2 and 9.1.3. During refueling, the fuel pool gate is open, and the equipment, reactor cavity and fuel pools function as a single volume. During this condition, the water in the reactor pressure vessel, reactor cavity pool, the equipment pool, and fuel pool (when gate is open) serves as the UHS. As indicated in PSAR Section 9.2.5.2, the management of this water inventory for use as UHS, related to IC pools and other flooded pools, will be described in the FSAR. This is acceptable because complete design information is not required to be submitted at the CP stage.

The NRC staff evaluated the applicant's compliance with the GDC listed above. GDC 2 requires capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods. The UHS is an integral part of the RB, and the pools are supported by the internal RB structure, which is classified as a Seismic Category I structure in accordance with GDC 2. PSAR Table 3A-2 provides classification of the RB which applies to the UHS pools, since they are an integral part and supported by RB. The PSAR Figure 9.2-3 shows the pool layout within RB. Based on the UHS location and configuration within the protected RB structure, the design should have capability to withstand the effects of natural phenomena and remain capable of performing its safety function. The NRC staff finds that the PSAR has identified relevant requirements and provided sufficient description of preliminary design criteria of the UHS to authorize construction. Therefore, the NRC staff finds that the design criteria presented in the PSAR are consistent with the requirements of GDC 2 and support issuance of the CP.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the service water subsystem design can be in conformance with GDC 5.

GDC 44, 45, and 46 require a system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink be provided. They also contain requirements related to testing and inspection. The UHS safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. The UHS is a passive safety feature and includes the IC pools, reactor cavity pool, fuel pool, and equipment pool. Reactor residual heat is transferred through the ICS to the isolation condenser pools during off-normal conditions, including anticipated operational occurrences (AOO) event sequences and DBA event sequences. GDC 44 is met by the heat removal capability of the ICS to transfer decay heat to the heat sink. GDC 45 and 46 are met because the pools are subject to testing and inspection as described in Section 9.2.5 of PSAR. Based on this capability, the NRC staff finds that the UHS is consistent with GDC 44, GDC 45 and GDC 46 and support issuance of the CP.

9.2.5.4 Conclusion

The NRC staff evaluated the UHS design description provided in PSAR Section 9.2.5 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that a UHS design constructed in accordance with these criteria could meet the requirements of GDC 2, 5, 44, 45 and 46 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

As described above, 10 CFR 50.34(a)(3)(iii) requires the PSAR submitted with the CP application to include “Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.” While PSAR contains time durations as basis for pool inventory, which is reasonable for CP review, the NRC staff will need to determine whether 72 hour and 7-day of water inventory is available within the pools for PCCS and IC performance during the OL review.

9.2.6 Condensate Storage and Transfer Subsystem

9.2.6.1 Introduction

This section discusses the condensate storage and transfer subsystem which is designed to provide condensate to the TB, RWB, and the RB for various systems. This subsystem is not safety-related and is not required to operate during or after an accident, with exception of containment isolation piping and valves. PSAR Figure 9.2-4, “Condensate Storage and Transfer Subsystem Simplified Diagram,” provides a simplified diagram of this subsystem.

9.2.6.2 Regulatory Evaluation

The NRC staff reviewed the design of the condensate storages and transfer subsystem in accordance with the review procedures in SRP Section 9.2.6, “Condensate Storage Facilities”, Revision 3. The following regulatory requirements apply:

- GDC 2, as it relates to the system’s capability to withstand the effects of natural phenomena, including earthquakes and tornadoes.
- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink.
- GDC 45, “Inspection of cooling water system,” as it relates to the design provisions to permit inspection of components and equipment.
- GDC 46, “Testing of cooling water system,” as it relates to the design provisions to permit operational testing of components and equipment.

- GDC 60, “Control of releases of radioactive materials to the environment,” as it relates to tanks and systems handling radioactive materials in liquids.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, “Contents of applications; technical information,” including (TN249):

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.”

- 10 CFR 50.35.

9.2.6.3 Technical Evaluation

The condensate storage and transfer subsystem is part of the LWM and consists of the condensate storage tank (CST), two transfer pumps, and distribution headers which supply condensate to the TB, RWB, and the RB for various systems. PSAR Figure 9.2-4 provides a simplified diagram showing the main supported systems or function flow paths from the CST.

The PSAR contains a list and reference to systems requiring use of the condensate storage and transfer subsystem. The system provides inventory for makeup, washing and filter backflushing which are further evaluated in the sections referenced in PSAR Section 9.2.6.2. PSAR Section 9.2.6.3 indicates that the condensate storage and transfer subsystem does not have direct liquid release paths to the environment.

The condensate storage and transfer subsystem (PSAR Figure 9.2-4) does contain containment penetration to allow water for a washdown connection within primary containment. As indicated in Table 6.2-2 “Process Line Isolation”, the liquid waste management system contains the inside and outside CIV related to the condensate storage subsystem, which are locked closed manual valves (Note 12 of PSAR Table 6.2-2). CIV are evaluated in Section 6.2.4 of this document. Table 6.2-4 indicates these CIV are locked closed during normal, hot shutdown and post-accident conditions.

The NRC staff evaluated the applicant’s compliance with the GDC listed above. The PSAR indicates that the design and layout of the condensate storage and transfer subsystem include provisions to ensure that any failure of the system will not adversely affect the function of SC1 components. In addition, the condensate storage and transfer subsystem is not required to operate during or after an accident. The NRC staff finds that the PSAR has identified relevant requirements and provided sufficient description of preliminary design criteria of the condensate storage and transfer subsystem to authorize construction. Therefore, the NRC staff finds that the preliminary design presented in the PSAR is consistent with the objective of GDC 2 and support issuance of the CP.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the service water subsystem design can be in conformance with GDC 5.

For condensate storage and transfer subsystem design, the applicant stated that GDC 44, 45, and 46 are not applicable to the condensate storage and transfer subsystem. SRP Section 9.2.6 Paragraph II.3 provides guidelines for how the condensate storage and transfer subsystem can meet GDC 44 related to performing the safety functions specified in SRP Section 9.2.6 Paragraphs II.3.A, II.3.B, II.3.C, II.4, and II.5. The NRC staff reviewed the system description of the condensate storage and transfer subsystem and found that the condensate storage and transfer subsystem does not have the safety function, as specified in SRP Section 9.2.6, to provide makeup water to safety-related cooling systems. Therefore, the NRC staff agrees with the applicant that GDC 44, 45, and 46 are not applicable for the condensate storage and transfer subsystem.

As indicated in PSAR, the condensate storage and transfer subsystems does not have a direct liquid release path to the environment and water is processed through the liquid waste management program described in Section 11.2. Therefore, the NRC staff has confidence that potential contamination of condensate storage and transfer system is addressed consistent with the requirements of GDC 60 to support issuance of CP.

9.2.6.4 Conclusion

The NRC staff evaluated the condensate storage and transfer subsystem design description provided in PSAR 9.2.6 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that a condensate storage and transfer subsystem design constructed in accordance with these criteria could meet the requirements of GDC 2, 5 and 60 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.2.7 Chilled Water Equipment

9.2.7.1 Introduction

The CWE is designed to provide chilled water for SC3 and SCN equipment. The CWE is non-safety related and not required to operate during or after an accident, with the exception of CIV and piping. PSAR Figure 9.2-5, “Chilled Water Equipment Simplified Diagram”, provides a simplified diagram of this system.

9.2.7.2 Regulatory Evaluation

The NRC staff reviewed the design of the CWE in accordance with the review procedures in SRP Section 9.2.7, Revision 0. Relevant regulations are:

- GDC 2, as it relates to the capability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without a loss of safety functions
- GDC 4, as to SSCs important to safety being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

- GDC 5, as it requires that SSCs important to safety not be shared among power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions.
- 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste (TN283).[3]

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.2.7.3 Technical Evaluation

The CWE is a closed-loop system that supplies chilled water to various AHU cooling coils and plant equipment coolers in the TB, RWB, RB, and CB. Heat absorbed by the CWE is rejected from the CWE chiller condensers to atmosphere. The CWE supplies chilled water to the containment cooling system (CCS) AHUs, which is described in Section 9.4.7 of PSAR. The CWE consists of two trains with redundant chiller and pumps containing a normally open cross tie to allow for the three active chillers to evenly share the heat load. During normal operation, three pumps are in service while the fourth is in standby mode. During a LOPP, one chiller and pump from each chiller train can be powered using backup power provided by the standby diesel generator. Chilled water is isolated from non-critical loads and supports the required component heat loads within the RB and CB.

The NRC staff evaluated the applicant's compliance with the GDC listed above. Per, PSAR Table 6.2-2 "Process Line Isolation", the chilled water supply and return piping contain an inside and outside CIV. As indicated in PSAR Section 9.2.7, the CWE does contain CIVs, which are described in Subsection 6.2.4 of PSAR.

---

[3] In accordance with 10 CFR 20.1002, 10 CFR Part 20 does not apply to CPAs. 10 CFR 50.40 describes considerations that will guide the Commission in determining that a CP will be granted. Per 10 CFR 50.40(a), one of those considerations is that "the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations...including the regulations in part 20...and that health and safety of the public will not be endangered." Decisions related to the design of the proposed facility affect an applicant's ability to meet the requirements of 10 CFR Part 20 in the future. Therefore, discussions of 10 CFR Part 20 in this section pertain to the NRC staff having reasonable assurance that a future operating reactor would be able to meet 10 CFR Part 20 considering the preliminary facility design.

The PSAR indicates that none of the equipment in the CWE, or cooled by the CWE, is required to mitigate DBAs. CWE piping or equipment include provisions to ensure that any failure of the system will not adversely affect the function of SC1 components. Therefore, the NRC staff finds that the preliminary chilled water system design is consistent with GDC 2 to support issuance of the CP.

PSAR Section 9.2.7.3 indicates that the CWE is designed for the plant normal operating environmental conditions specified for its location within the TB and RB. In addition, none of the equipment in the CWE, or cooled by the CWE, is required to mitigate DBAs. The Chilled Water System is a moderate energy piping system evaluated in PSAR Section 3.6.1 and designed to not adversely affect the function of SC1 components. Based on this, the NRC staff finds that the preliminary design is consistent with GDC 4 and support issuance of the CP.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the service water subsystem design can be in conformance with GDC 5.

Some of the heat exchangers cooled by the CWE are in radiologically controlled areas. Consistent with 10 CFR 20.1406, programs are in place to minimize the spread of contamination as discussed in PSAR Section 12.1. The NRC staff evaluates these programs in chapter 12 of this report.

9.2.7.4 Conclusion

The NRC staff evaluated the CWE description provided in PSAR 9.2.7 and concluded that the applicant has proposed adequate description for the system. The NRC staff has confidence that a CWE design constructed in accordance with these criteria could meet the requirements of GDC 2, 4 and 5 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.2.8 Isolation Condenser Pools Cooling and Cleanup System

9.2.8.1 Introduction

This section discusses the ICC system. The ICC system is used to maintain isolation condenser (IC) pool operating parameters, such as temperature, level, pool inventory, and water quality. The ICC maintains the water in the IC pool complex located in the RB which consists of the IC inner pools, IC outer pools, and the filter/storage pool (shown in PSAR Figure PSAR 9.2-6).

PSAR Section 9.2.8 provides description of the components within the pools and supporting function of the ICC pools and filter/storage pools. Operation of the isolation condensers (ICS) within the ICC pools is discussed in PSAR Section 6.3 and description of the UHS pools is contained in PSAR Section 9.2.5.

9.2.8.2 Regulatory Evaluation

The NRC staff determined that no current SRP is directly applicable to the ICC, which provides function to maintain IC pool parameters. Since it is considered part of the isolation condenser function, and because it is part of the UHS, the NRC staff reviewed the ICC in accordance with

SRP Section 9.2.5, Revision 3. The NRC staff's acceptance of the design is based on compliance with the following requirements:

- GDC 2, as to capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink
- GDC 45, "Inspection of cooling water system," as it relates to the design provisions to permit inspection of components and equipment.
- GDC 46, "Testing of cooling water system," as it relates to the design provisions to permit operational testing of components and equipment.
- Although not listed in SRP 9.2.5, GDC 4 is applicable due to the piping and tanks within this system.
- GDC 4 as to effects of missiles inside and outside of containment, effects of pipe whip, jets, environmental conditions from high- and moderate-energy line breaks, and dynamic effects of flow instabilities and attendant loads (i.e., water hammer) during normal plant operation as well as upset or accident conditions.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.2.8.3 Technical Evaluation

The NRC staff evaluated the sufficiency of the Clinch River preliminary design features for systems and components, as described in PSAR Section 9.2.8, for the issuance of a CP using the applicable guidance and acceptance criteria from Section 9.2.5, "Ultimate Heat Sink," of NUREG-0800 (NRC 2021-TN8013). Staff acceptance of the ICC is based on meeting the requirements of GDC 2, 5, 44, 45, and 46.

The PSAR indicates the Safety Category 1 functions related to pool inventory management are:

- The water in the IC Pools is the UHS, which removes heat from the reactor core through the ICS during adverse conditions and DBAs. The ICS functions are described in Section 6.3.

- The IC Pools Atmospheric Vents minimize pressurization of the IC Pool compartments by providing a low-resistance flow path for steam to be released to the environment during IC deployment.
- The unidirectional pool makeup conduits allow makeup water to flow from the IC outer pools into the IC inner pools following extended IC deployment with significant loss of pool water inventory due to boiloff.

The PSAR describes the ICC Safety Category 3 function to provide long-term makeup to the IC and Safety Category N function to maintain pool operating parameters such as temperature, level, and water quality.

The component requirements for the IC inner and outer pools are defined in PSAR Section 9.2.8.2 and listed below:

- The smaller IC inner pool (IC Inner Pool A) and the full volume of the outer pools shall be capable of cooling the reactor for 72 hours after shutdown without replenishment.
- The larger IC inner pool (IC Inner Pool B + C) and the full volume of the outer pools shall be capable of cooling the reactor for 7 days after shutdown without replenishment.

After 7 days, supplemental water can be provided through the IC Pool FLEX/EME connection.

The ICC consists of two cooling trains, equipped with a centrifugal pump and heat exchanger to transfer energy from the IC Pools to the PCW (PSAR Section 9.2.2). The ICC also includes a demineralizer to remove impurities from the IC pool and filter/storage pool water. Vents at high locations in the piping system ensure pipes are filled with process water and eliminate air pockets. The ICC draws water for subsequent cooling and/or purification from the suction surge tank located in the IC Outer Pool B + C. This design prevents IC outer pool water level from falling below the top of the suction surge tank (and subsequently, normal minimum IC pool water level) in the event of a non-isolable pipe break below the IC pools. Processed water is returned to the IC inner pools through a piping network that ends at distribution spargers mounted near the ICs and the bottom of the filter/storage pool.

The ICC contains instrumentation designed to measure local process flow variables: water level, temperature, pressure, differential pressure, and conductivity. Levels and temperatures in the IC pools, equipment pool, reactor cavity pool, and fuel pool will be monitored and maintained within administrative and technical specification limits, as indicated in Section 9.2.5.4 of PSAR. Although 10 CFR 50.34(a)(5) requires an identification and justification for the selection of those variables, conditions, or other items which are determined as the result of the preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, Technical Specifications are not provided in the PSAR. This information is not the sort that falls under those requirements. Therefore, the NRC staff will further evaluate technical specification limits during OL stage. PSAR Table 3A-1 lists the preliminary classification of the ICC system as SC1 for piping and components impacting IC pools function, as indicated above, and SC3 for components related to filter/storage pool. Other ICC components used to maintain pool inventory during normal operation are classified as SCN.

The following SC1 design functions apply to the designed safety features of the IC pools including: (1) the suction surge tank, return guard tank (including enclosed piping and anti-siphon device), which serves as the primary physical interface with the RB IC pool structure, (2) the IC Pools atmospheric vents, which minimize pressurization of the IC pool compartments, (3) the unidirectional pool makeup conduits, which allow makeup water to flow from the outer pools to the IC inner pools during off-normal conditions, and (4) IC pool isolation valves, which are attached to the pool penetrations that are contained within the suction surge tank and return guard tank.

GDC 2 requires capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods. The IC pools are an integral part of the RB, and supported by the internal RB structure, which is classified as a Seismic Category I structure in accordance with GDC 2. The SSC components that interface or perform supporting functions are designed as seismic I components, as shown in PSAR Table 3A-1. GDC 2 is addressed for the non-safety portions of the system performing normal operating functions based on the design and layout of the ICC design having no adverse interaction with the SC1 Pools or its connected piping. Based on UHS location within the RB providing protection against external hazards and ICC components classification, the NRC staff finds the design meets GDC 2.

PSAR Section 9.2.8.3 indicates that the IC pools serviced by the ICC are located in seismic Category I structure to protect SSCs 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. The applicant also indicates the Suction Surge Tank and the Return Guard Tank located within the IC pools are evaluated for environmental and dynamic effects of design basis events. The NRC staff finds that the PSAR has provided sufficient description of preliminary design criteria of the ICC to authorize construction. Therefore, the NRC staff finds that the design criteria presented in the PSAR are consistent with the requirements of GDC 4 and support issuance of the CP.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the service water subsystem design can be in conformance with GDC 5.

GDC 44, 45, and 46 require a system to transfer heat from SSCs important to safety, to an ultimate heat sink shall be provided. They also impose certain requirements related to inspections and testing. The ICC maintains the ICC pool inventory under normal operation and ICC inventory makeup is not required to support the UHS function during accident conditions. Section 9.2.8.3 of PSAR indicates that the ICC is designed to permit periodic inspection and testing to ensure the integrity and capability of the system. Isolation valves are provided near the IC pool penetrations, and the ICC piping component configuration protects against inadvertent draining of IC pools, per PSAR Section 9.2.8.2. For ICC, the applicant stated that GDC 44, 45, and 46 are not applicable to the ICC. SRP Section 9.2.5 provides guidelines for how UHS can meet GDC 44 related to performing the safety functions. The NRC staff reviewed the system description of the ICC and found that the system does not have an applicable GDC 44 safety function to provide makeup water to safety-related cooling systems during accident conditions. Therefore, the NRC staff agrees with the applicant that GDC 44 are not applicable for the ICC function during normal operation. However, additional reviews of GDC 45 and 46 requirements for inspection and test programs for the safety related tanks and features of ICC are needed during OL stage. The NRC staff does not need to reach a conclusion about GDCs 45 and 46 at this time because, given the current state of the design, the NRC staff cannot determine whether they are applicable.

9.2.8.4 Conclusion

The NRC staff evaluated the condensate storage and transfer subsystem design description provided in PSAR Section 9.2.8 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that an ICC design constructed in accordance with these criteria could meet the requirements of GDC 2, 4 and 5 of Appendix A of 10 CFR Part 50. Additional discussion of GDC 44, 45 and 46 are included above. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

As described in PSAR Section 9.2.5 above, 10 CFR 50.34(a)(3)(iii) requires CPAs to include a PSAR that "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety." While PSAR contains time durations as basis for pool inventory, water volumes will be needed to verify the 72 hour and 7-day of water inventory, which is reasonable for CP review, the NRC staff will need to determine whether 72-hour and 7-days of water inventory is available within the pools for IC and PCCS performance during the OL review.

9.2.9 Normal Heat Sink

9.2.9.1 Introduction

This section discusses the NHS. The NHS system provides heat rejection for the circulating water system (CWS) and service water (SW) subsystem during normal operation and shutdown.

The heat from the SW subsystem (Section 9.2.1) is transferred to the NHS. SW provides cooling water to the PCW system (Section 9.2.2) heat exchangers.

9.2.9.2 Regulatory Evaluation

The NRC staff reviewed the design of the Normal Heat Sink in accordance with SRP Section 9.2.5 (Revision 3). This SRP is typically for UHS of light water reactors to design UHS with safety related heat sink capability during normal, shutdown and accident conditions. For the proposed design, plant cooling during normal operation is by NHS and the IC pools act as the UHS for accident conditions. Therefore, SRP 9.2.5 is chosen for the review of NHS design. The NRC staff's acceptance of the design is based on compliance with the following requirements:

- GDC 2, as to capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods.
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions
- GDC 44, as it relates to transferring heat from SSCs important to safety to a heat sink
- GDC 45, "Inspection of cooling water system," as it relates to the design provisions to permit inspection of components and equipment
- GDC 46, "Testing of cooling water system," as it relates to the design provisions to permit operational testing of components and equipment

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):

– 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

– 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

– 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."

- 10 CFR 50.35.

9.2.9.3 Technical Evaluation

The NRC staff evaluated the sufficiency of the Clinch River preliminary design features for systems and components, as described in PSAR Section 9.2.9, for the issuance of a CP using the applicable guidance and acceptance criteria from Section 9.2.5, "Ultimate Heat Sink," of NUREG-0800 (NRC 2021-TN8013). Staff acceptance of the NHS is based on meeting the requirements of GDC 2, 5, 44, 45, and 46.

The NHS system includes the cooling tower, the intake structure, makeup water pumps, the holding pond (also known as cooling tower blowdown pond), the diffuser, and other associated equipment. The main function of the NHS is to support heat rejection for the CWS and SW subsystem during normal operation and shutdown. The NHS cooling tower obtains water inventory from Clinch River using four makeup water pumps and a simplified diagram of the NHS system is provided in Figure 9.2-7. The NHS system also provides water for the DWS.

During normal operation, the NHS system transfers heat from plant equipment to the environment. The NHS is not used to cool safety-related SSCs and does not function as the UHS to mitigate accidents. In the event of an accident or if the NHS system is not available, heat removal will be performed by the UHS. The UHS is described in PSAR Section 9.2.5 and includes the IC pools, reactor cavity pool, fuel pool, and equipment pool. Specific capacity, flows, and other details of the NHS are not provided in the PSAR and will be reviewed during the OL review.

The PSAR indicates that the design and layout of the NHS system include provisions to ensure that any failure of the system will not adversely affect the function of SC1 components. PSAR Table 3A-1 defines classification of the normal heat sink as SC3 for components associated with heat sink function for CWS and SW, and SCN for all other components. In addition, the NHS system is not required to mitigate a DBA. Therefore, the NRC staff finds that the preliminary NHS design is consistent with GDC 2 to withstand the effects of natural phenomena and support issuance of the CP.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the NHS design can be in conformance with GDC 5.

In PSAR Section 9.2.9.3, the applicant stated that there are no specific requirements implemented for GDC 44, 45, and 46 for the NHS. SRP Section 9.2.5 provides guidelines for how UHS can meet GDC 44 related to performing the safety functions, but the NHS does not perform any safety function. Similarly, GDC 45 and 46 require inspection and testing to ensure the safety related SSC can perform safety function. The NRC staff reviewed the system description of the NHS and found that the system does not have an applicable safety function and is not required to safely shutdown plant during accident conditions. Therefore, the NRC staff agrees with the applicant that GDC 44, 45 and 46 are not applicable for the NHS function to support issuance of CP.

PSAR Section 9.2.9.4 indicates NHS contains input streams to the NHS system from potentially radioactive systems that are monitored for radioactivity prior to entering the NHS.
Section 9.2.9.3 of the PSAR describes a liquid waste management discharge path to CWS for dilution and release to environment which is monitored and isolated upon high radiation. In addition, PSAR Section 9.2.1.3 describes radiation monitoring installed on SW subsystem piping downstream of the PCW heat exchanger to detect leakage from the PCW to the SW subsystem, which could discharge into the NHS. The NRC staff evaluates process radiation monitoring in Section 11.5 of this report.

9.2.9.4 Conclusion

The NRC staff evaluated the NHS design description provided in PSAR Section 9.2.9 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that an NHS design constructed in accordance with these criteria could meet the requirements of GDC 2 and 5 of Appendix A of 10 CFR Part 50. Additional discussion of GDC 44, 45 and 46 are included above. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.2.10 Refueling Water Storage and Cleanup Subsystem

9.2.10.1 Introduction

This section discusses the Refueling Water and Cleanup Subsystem. This system is designed to provide and store water inventory for use during refueling outage activities. This subsystem is not safety-related and is not required to operate during or after an accident. PSAR Figure 9.2-8, “Refueling Water Storage and Cleanup Subsystem Simplified Diagram”, provides a simplified diagram of this subsystem.

9.2.10.2 Regulatory Evaluation

The NRC staff reviewed the design of the Refueling Water Storage and Cleanup Subsystem in accordance with the review procedures in SRP Section 9.2.6 “Condensate Storage Facilities”, Revision 3, since this system does not have a specific SRP. The NRC staff finds this SRP is applicable because the system Clinch river system performs similar function of the condensate storage facility. The following specific regulatory requirements apply:

- GDC 2, as it relates to the system’s capability to withstand the effects of natural phenomena, including earthquakes and tornadoes.
- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions.
- GDC 60, as it relates to tanks and systems handling radioactive materials in liquids.

  The NRC staff also finds the following regulations applicable to the design:
- 10 CFR 50.34, “Contents of applications; technical information,” including (TN249):

– 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

– 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

– 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."

- 10 CFR 50.35.

9.2.10.3 Technical Evaluation

The Refueling Water Storage and Cleanup Subsystem is part of the LWM and consists of the refueling water storage tank (RWST), two transfer pumps, a filter and associated piping. The main function is to transfer, hold, and filter water used to fill the reactor cavity pool and other connected systems. The system also has the function of confinement of water that contains radioactivity. PSAR Figure 9.2-8 provides a simplified diagram showing the main system components and flow paths from the RWST. The RWST is located in the turbine building.

The PSAR indicates that the design and layout of the Refueling water storage and cleanup subsystem include provisions to ensure that any failure of the system will not adversely affect the function of SC1 components. In addition, this subsystem is not required to operate during or after an accident. The RWST is classified as non-seismic and the applicant should address any flooding concerns. The NRC staff discussed tank flooding in audit and applicant indicated the failure analysis is not finalized and will be confirmed in the FSAR considering final piping layout and internal flood analysis. Therefore, the NRC staff finds the preliminary design is consistent with GDC 2 to withstand the effects of natural phenomena and support issuance of the CP and will review potential flooding during the OL review.

The Clinch River Nuclear (CRN-1) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the Refueling Water and Cleanup Subsystem design can be in conformance with GDC 5.

The PSAR defines the purpose of the subsystem as to hold the reactor cavity pool volume to support refueling outage activities. The PSAR indicates that this subsystem contains radiation and has function for confinement of water that contains radioactivity. PSAR Section 9.2.10.3 indicates that the subsystem does not have direct liquid release paths to the environment. To protect against failure and mitigate leakage, the RWST contains overflow monitoring and alarms with the ability to direct overflow to equipment and floor drain system (EFS). Walls/curb are also provided around the RWST to collect leaks and prevent release to groundwater. The RWST also has option to transfer water to be processed through the LWM. Therefore, the NRC staff finds that the preliminary design presented in the PSAR is consistent with GDC 60 and support issuance of the CP

9.2.10.4 Conclusion

The NRC staff evaluated Refueling Water and Cleanup Subsystem design description provided in PSAR Section 9.2.10 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that a Refueling Water and Cleanup Subsystem design constructed in accordance with these criteria could meet the requirements of GDC 2, 5 and 60 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.3 Process Auxiliaries

9.3.1 Plant Pneumatics Systems

9.3.1.1 Introduction

The plant pneumatics system (PPS) consists of the service air (SA) subsystem and the instrument air (IA) subsystem. The PPS provides dried and filtered compressed air to valve actuators, instrument control functions, general instrumentation and valve services, and compressed air for maintenance activities. PSAR Section 9.3.1 indicates that the PPS does not have a Safety Category 1 or 2 function. The system performs the following Safety Category 3 functions:

- provide a supply of service air or instrument air to other systems for tank sparging, filter/demineralizer backwashing, air-operated valves, and other services
- provide distribution of pressurized nitrogen to other systems in the containment, where the atmosphere is inerted during operation

The PPS piping is used to connect the containment inerting system (CIS) to pneumatic valves and other compressed gas users inside the containment. The CIS is described in PSAR Section 9.3.6. The NRC staff reviewed the CIS in Section 9.3.6.

9.3.1.2 Regulatory Evaluation

A CP must meet the requirements of 10 CFR 50.34(a) (TN249). This requires the CP applicant to provide information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety. The NRC staff reviewed the PPS and CIS in accordance with SRP Section 9.3.1, Revision 2, “Compressed Air System,” issued March 2007, which provides guidance to the NRC staff on the relevant regulatory requirements for this area of review, summarized below, as well as the review interfaces with other SRP sections:

- GDC 1, as it relates to important-to-safety SSCs designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed.
- GDC 2, as it relates to important-to-safety SSCs being designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.
- GDC 5, as it relates to the sharing of important-to-safety SSCs among nuclear power units.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, “Contents of applications; technical information,” including:
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

– 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

– 10 CFR 50.34(a)(3)(iii) requires that the PSAR include “Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.”

- 10 CFR 50.35.

9.3.1.3 Technical Evaluation

As indicated in the previous section of this report, the regulatory requirements for a CPA are presented in 10 CFR Part 50.33 and 50.34(a). As part of the application, the PSAR Section 9.3.1 provides the specific design criteria applicable to the PPS.

The NRC staff reviewed PSAR Section 9.3.1, in accordance with the guidance provided in SRP Section 9.3.1. The applicant stated that the PPS is composed of the instrument and service air subsystems which the applicant indicated have no Safety Category 1 or 2 function. The instrument air subsystem penetrates the containment that connects the PPS and CIS to compressed gas users inside containment. During operation, the air supply into containment is isolated, and CIS provides nitrogen gas to users inside containment. During plant shutdown, the nitrogen supply is isolated from the containment. The application indicates that this is done to ensure that during operation, the containment environment does not pose an explosion or fire risk, and that during outages the containment does not turn into a confined space or pose an asphyxiation risk to plant personnel.

In PSAR Section 9.3.1.2 the applicant indicates that the PPS instrument air subsystem is designed and maintained to the standards of American National Standards Institute (ANSI)/International Society of Automation (ISA) 7.0.01, “Quality Standard for Instrument Air.”

The NRC staff’s guidance in SRP 9.3.1 makes reference to ANSI/ISA-S7.3-R1981, which provides air quality criteria to ensure that instrument air system and the equipment supplied from it will reliably perform their intended safety functions. ANSI/ISA-S7.3-R1981 has been withdrawn and replaced by ANSI/ISA 7.0.01-1996. The NRC staff compared the air quality criteria identified in ISA 7.0.01 against the criteria identified in ANSI/ISA-S7.3-R1981 and determined that both standards provide comparable criteria for dew point, allowable oil content, contaminants, scope, and testing requirements, without introducing a significant relaxation in requirements.

ANSI/ISA-7.0.01 relaxes the maximum allowable particle size in the instrument air system from 3 micrometers (μm) to 40 μm or a more limiting particle size, if the pneumatic devices may require instrument air with a more limiting maximum particle size. The particle size is determined by the pneumatic devices that are serviced by the PPS and controlled by the replaceable filter media used, which is an operational parameter, not required to be reviewed for a construction license. During the review of the FSAR, the NRC staff will confirm that the maximum allowable particle size is consistent with the air quality requirements of safety related air-operated SSCs.

Based on the above, the NRC staff evaluated the design criteria identified in ANSI/ISA-7.0.01 and determined that it is acceptable for the CRN Site to design the instrument air subsystem following this standard.

The portion of the PPS that penetrates the containment building and associated supports are designed to Seismic Category I, the remaining of the PPS is designed as non-seismic, except where a component failure could adversely interact with Seismic Category I SSCs during a safe shutdown earthquake (SSE). In PSAR Section 3.1.1.2, the applicant indicates that "SSCs whose failure during a seismic event could prevent a Seismic Category I SSC from performing its Safety Category 1 functions are designed to ensure such failure does not occur." The NRC staff evaluated the system description provided in PSAR Section 9.3.1 and determined that the applicant has identified adequate design criteria and seismic design for the PPS. This classification is consistent with the recommendation provided in SRP Section 9.3.1 as adequate design to conform with the requirements of GDC 1, 2, and 4.

GDC 5 requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units. The CRN Site is a single-unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the PPS design can be in conformance with GDC 5.

In PSAR Section 9.3.1.3 the applicant indicates that the PPS considered the requirements of 10 CFR 50.63, as it relates to loss of all alternating current power. The PPS is not credited to achieve safe shutdown, it does not support safety category functions pertaining to maintaining the ability to actuate or control equipment necessary for core cooling and decay heat removal or maintaining containment integrity following a SBO.

The NRC staff notes that this information is not required to meet the applicable regulatory requirements for a CP under 10 CFR 50.33 through 50.35 and 10 CFR Part 50, Appendix A. The NRC staff acknowledges that TVA included this information in its PSAR; however, the NRC staff has not evaluated this information as part of its safety review to support the issuance of the CP. Accordingly, the NRC staff does not take a position on the information presented in this section related to compliance with the requirements of 10 CFR 50.63.

9.3.1.4 Conclusion

The NRC staff evaluated the PPS design description provided in PSAR 9.3.1 and concluded that the applicant has proposed adequate design criteria for the system. The NRC staff has confidence that PPS constructed in accordance with these criteria could meet the requirements of GDC 1, 2, and 5 of Appendix A of 10 CFR Part 50. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

The NRC staff will review the FSAR to confirm that the applicant's specified maximum particle size is consistent with the air quality requirements associated with safety-related air operated SSC.

9.3.2 Process Sampling System

9.3.2.1 Introduction

This section identifies and describes the process sampling system (PS) design that will be used to obtain liquid and gaseous samples and determine their chemical and radiochemical conditions by measurement and analysis. PSAR Section 9.3.2, “Process Sampling System,” states that the PS is part of the process and radiation and environmental monitoring system (PREMS) and describes that automated and local facilities permit samples of process streams and components to be taken, which provides analytical information required for monitoring plant and equipment performance.

The PS provides online chemistry monitoring for the following systems, and these samples are routed to an automated sample panel:

- control rod drive (CRD) purge water
- shutdown cooling system
- reactor water cleanup system
- FPC
- LWM
- condensate and feedwater heating system
- condensate filters and demineralizers system
- main condenser and auxiliaries

The PS provides local and grab sampling equipment for the following systems:

- ICC system
- solid waste management system (SWM)
- condensate and feedwater heating system
- PCW
- EFS

Sampling racks and electronic modules are maintained per industry standards or operational guidelines to ensure reliability. Routine testing and calibration are part of the maintenance plan, and instruments are re-calibrated after any work that may affect their accuracy. The system is designed to allow easy access to sensors, instruments, and panels for maintenance, with instrument modules optimized for calibration and troubleshooting.

9.3.2.2 *Regulatory Evaluation*

In accordance with 10 CFR 50.34 (TN249), “Contents of Applications; Technical Information”, GDC 13, 14, 60, 63, 64, and PDC 26, the plant must be designed to monitor certain parameters for hazardous conditions.

The following NRC regulations, as interpreted in SRP Section 9.3.2, Revision 3, are relevant for this review:

- The NRC considered 10 CFR 20.1101(b), in part, that sound radiation protection principles will be used as part of the design of the proposed facility to achieve occupational doses and doses to members of the public that are ALARA.[4]

- In GDC 1, the NRC requires, in part, that the PS and components shall be designed in accordance with standards commensurate with the importance of their safety functions.

- In GDC 2, the NRC requires, in part, that the PS shall have the ability to withstand the effects of natural phenomena.

- In GDC 13, the NRC requires, in part, that there shall be the capability to monitor variables that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary.

- In GDC 14, the NRC requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. As discussed in the SRP guidance for evaluating chemistry controls that support reactor coolant pressure boundary (RCPB) integrity, applicants should describe its capability to sample chemical species that could affect the RCPB.

- The applicant is proposing to implement a design-specific principal design criterion (PDC) 26 in lieu of GDC 26. The NRC staff evaluated the acceptability of PDC 26 in Chapter 4.6 of this report.
- In GDC 60, the NRC requires, in part, that the PS shall be capable of controlling the release of radioactive materials to the environment.
- In GDC 63, the NRC requires, in part, that there shall be the capability provided to detect conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems.
- 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.

SRP Section 9.3.2, Revision 3, "Process and Post-Accident Sampling System," provides the criteria that are acceptable to meet the relevant regulatory requirements as well as interfaces

[4] In accordance with 10 CFR 20.1002, 10 CFR Part 20 does not apply to CPAs. 10 CFR 50.40 describes considerations that will guide the Commission in determining that a CP will be granted. Per 10 CFR 50.40(a), one of those considerations is that "the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations...including the regulations in part 20...and that health and safety of the public will not be endangered." Decisions related to the design of the proposed facility affect an applicant's ability to meet the requirements of 10 CFR Part 20 in the future. Therefore, discussions of 10 CFR Part 20 in this section pertain to the NRC staff having reasonable assurance that a future operating reactor would be able to meet 10 CFR Part 20 considering the preliminary facility design.

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

- RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," Revision 1, describes acceptable methods that may be used for measuring, evaluating, and reporting radioactive material, and assessing and reporting the public dose (NRC 2024-TN13182).
- RG 1.26, "Quality Gorup Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 6, provides guidance for alternative quality classification systems for components in light-water reactor (LWR) nuclear power plants (NRC 2021-TN12803.
- RG 1.29, "Seismic Design Classification for Nuclear Power Plants," Revision 6, provides acceptable methods for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the safe-shutdown earthquake (SSE) (NRC 2021-TN12804).

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including:
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.3.2.3 Technical Evaluation

The NRC staff reviewed the Clinch River preliminary design provisions for the process sampling system in accordance with the guidance and requirements listed above. The results of the NRC staff's review are provided below.

The PS monitors process and effluent streams for potentially hazardous contamination such as radioactivity, noble gases, air particulates, and halogens, consistent with 10 CFR Part 20 (TN283) and 10 CFR Part 50 related to monitoring and reporting of radioactive material in effluents and environmental media, solid radioactive waste shipments, and the public dose that results from licensed operation of a nuclear power plant. The NRC staff determined that the PS conforms to RG 1.21, because for processes requiring continuous or frequent monitoring, sample tubing routes process fluid to automated sample panels equipped with conditioning components for pressure, temperature, and flow control. These panels support online analyzers for key parameters such as conductivity, dissolved gases, total organic carbon, and silica. Grab sample taps are included for infrequent sampling, analyzer verification, or when online systems are unavailable. Test connections are also provided for calibration or temporary instrumentation. Infrequent or diagnostic samples are typically collected at local grab sample stations, which may also include conditioning equipment as needed. The PS equipment is distributed throughout the plant, with automated panels located near the chemistry laboratory for efficiency, and grab sample stations positioned close to their respective process streams.

PSAR Table 3A-1, “Preliminary BWRX-300 Component Classification List,” indicates that the RG 1.26 Quality Group for the non-pressure boundary sampling equipment and the non-contaminated pressure boundary sampling equipment is not applicable because the components do not perform a SC1 pressure boundary function nor are part of systems or portions of systems that contain or may contain radioactive material. Additionally, the pressure boundary sampling equipment that may contain radioactive material is classified as Quality Group D and conforms to Regulatory Position C.4 in RG 1.26. The seismic category for all PS components is classified as non-seismic. The NRC staff reviewed the applicant’s seismic design and quality group classification of sampling lines and components for the PS and its conformance to the classification of the system to which each sampling line and component is connected and finds that is consistent with RG 1.26 and RG 1.29. Therefore, the NRC staff finds that the CRN-1 proposed design is consistent with the requirements of GDC 1 and 2, as it relates to the process sampling system.

PSAR Section 9.3.2 describes that the PS design provides the capability to sample the reactor coolant system and associated auxiliary systems, therefore the NRC staff finds that the PS is consistent with the requirements of GDC 13 to monitor variables that can affect the fission process, the integrity of the reactor core, and the RCPB during normal operation.

The PS monitors parameters that may indicate leakage from the RCPB. In addition, fission product samples are manually collected from the containment and analyzed to detect potential reactor coolant leakage within the containment area. By providing the capability to sample and analyze reactor system coolant samples to verify key parameters are within prescribed parameters, the NRC staff finds that the PS is consistent with GDC 14 to monitor variables that can affect the RCPB and to ensure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

For the CRN-1 proposed design, the BIS is not relied upon for normal operation, including anticipated operational occurrences, however, the BIS is provided as a defense-in-depth measure. PSAR Section 9.3.2 states that the design of the BIS allows for periodic sampling and surveillance of the boron concentration. Therefore, the NRC staff finds that the PS is consistent with the requirements of PDC 26, to control the rate of reactivity changes by providing the capability to sample the boron concentration.

As discussed above, the PS collects representative liquid and gaseous samples for analysis, providing essential data for monitoring plant and equipment performance. Consequently, the PS may contain radioactive material. The PS equipment that is suspected of containing radioactive substances is designated as SC3 to ensure proper confinement of radioactivity. PSAR Section 9.3.2 describes the PS design features, which conform to RG 1.21, that help control the release of radioactive material to the environment, which enhances safety. Based on these design features, the NRC staff finds that the PS is consistent with GDC 60, to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation.

The PS design includes design features to monitor conditions that may result in excessive radiation levels in the fuel pool and radioactive waste systems. The PS also includes sampling capability for LWM and SWM systems, which enable analyses to be performed to detect conditions related to fuel storage and radioactive waste systems that could result in excessive radiation levels and excessive personnel exposure. The NRC staff's review of LWM and SWM is discussed in Section 11.2 and 11.4, respectively. Based on the design of the PS, the NRC staff finds that the PS is consistent with GDC 63 to detect conditions that may result in excessive radiation levels in fuel storage and waste storage by providing the capability to monitor the fuel pool and sample LWM and SWM for radioactivity.

The PS does not provide monitoring of the containment atmosphere for radioactivity. However, the containment monitoring system (CMon), which is another subsystem of PREMS does provide this capability. Therefore, GDC 64 for monitoring radioactive releases in the containment is not applicable to PS. The NRC staff's review of CMon and consistency with GDC 64 is discussed in Section 6.8 of this safety evaluation report.

The objective of the ALARA requirements is to ensure that reasonable efforts associated with the planning, design, and operation of the PS maintain exposures to radiation as far below the limits of 10 CFR Part 20 as is reasonably achievable. PSAR Section 9.3.2.3 confirms that the PS design features, and configuration, considers ALARA program goals and objectives with regard to minimizing dose contamination. Station layout and design features are developed in support of reducing the potential doses to personnel who operate, service, or inspect the PS. Therefore, the NRC staff finds that the PS design considers the use of engineering controls in maintaining doses ALARA and, based on that consideration, there is reasonable assurance that 10 CFR 20.1101(b) will be met consistent with 10 CFR 50.40(a).

9.3.2.4 Conclusion

The NRC staff review concludes that the applicant's preliminary process sampling system design for CRN meets 10 CFR 50.34(a)(3) and is consistent with the NRC regulations as set forth in GDC 1, GDC 2, GDC 13, GDC 14, PDC 26, GDC 60, GDC 63, to monitor plant parameters for hazardous conditions. The NRC staff concludes this based on the applicant addressing the applicable GDCs listed in SRP Section 9.3.2 and with regard to GDC 60 and 50.34(a), because the applicant used acceptable methods in accordance with RG 1.21. Further, the NRC staff concludes that, for the reasons stated above, GDC 64 is not applicable to the PS and there is reasonable assurance that 10 CFR 20.1101(b) will be met, consistent with 10 CFR 50.40(b). Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.3.3 Equipment and Floor Drain System

9.3.3.1 Introduction

This section discusses the EFS. The EFS is designed to collect liquid throughout the plant, including radioactive or potentially radioactive liquid waste generated within RB and transfer these wastes to an appropriate disposal or collection system.

9.3.3.2 Regulatory Evaluation

The NRC staff reviewed the proposed design for EFS in accordance with NUREG-0800 (SRP) Section 9.3.3 (NRC 2021-TN8013), "Equipment and Floor Drainage System". The NRC staff's review is based on compliance with the following requirements:

- GDC 2, as to important to safety system portions being capable of withstanding the effects of natural phenomena.

- GDC 4, as to capability to withstand the effects of and to be compatible with the environmental conditions (flooding) of normal operation, maintenance, testing, and postulated accidents (pipe break, tank ruptures).
- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.
- GDC 60, as to suitable control of the release of radioactive materials in liquid effluent, including anticipated operational occurrences. This criterion applies as the drains usually consist of two subsystems, radioactive and nonradioactive. The inadvertent transfer of radioactive waste to the nonradioactive portion of the system could result in radioactive releases to the environment.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.3.3.3 Technical Evaluation

The NRC staff evaluated the sufficiency of the Clinch River preliminary design features for systems and components, as described in PSAR Section 9.3.3, for the issuance of a CP using the applicable guidance and acceptance criteria from SRP Section 9.3.3. Staff acceptance of the EFS is based on meeting the requirements of GDC 2, 4, 5, and 60.

The EFS consists of drain piping, collection sumps, sump pumps, discharge piping, and instrumentation to facilitate the collection, removal, and measurement of liquid waste in the plant. During a Design Basis Accident (DBA) and assuming the unavailability of normal power, EFS does not operate, nor is it required to operate.

The EFS is comprised of the following four types of sump collection systems which are described in PSAR Section 9.3.3:

- general EFS
- pressurized sump to the general floor drain system
- oily waste sumps
- dedicated potentially clean drain systems

Each of the above subsystems has its own sump, pumps, isolation valves, and instrumentation and piping. Clean, oily, and contaminated waste is collected separately via the EFS.

General EFS collects the liquid waste generated throughout the plant. These drains include floor drains, equipment drains, and process drains which collect and route to LWM for appropriate processing. Any liquid waste collected in radioactive drains is treated as high conductivity waste (HCW). Processing of waste in the LWM is described in PSAR Section 11.2. PSAR Figure 9.3-2 provides simplified diagram for the normal sump for general EFS collection.

General floor drain systems are included in the basement floor of the RB, including a pressurized sump vessel which is used to monitor containment leakage. The PSAR indicates the piping upstream of the pressurized sump penetrates the containment, necessitating double isolation valves for containment isolation events which are discussed in PSAR Section 6.2.4. The pressurized containment sump level is monitored to measure and trend containment leakage during normal operation. Excessive containment leakage is detected by the containment sump level changes. As described in PSAR Section 5.2.5, the containment sump collects unidentified leakage from the floor drain and condensate from containment atmosphere coolers. PSAR Figure 9.3-3 provides a simplified diagram for the pressurized sump.

Oily waste sumps are provided at the base of the turbine electro-hydraulic control unit, lube oil skid, and diesel generator skids to collect and identify leakage and proper disposal. PSAR Figure 9.3-4 provides simplified diagram for the oily waste sump.

Dedicated potentially clean subsystems are provided as a dedicated drainage system because the water is potentially clean. Systems using these drains are the plant pneumatics system as discussed in Section 9.3.1, and PCW system described in PSAR Section 9.2.2.
PSAR Section 9.3.3.3 and Table 1.9-9 indicates EFS complies with GDC 2. PSAR Section 9.3.3.3 further indicates that EFS compliance follows the plant-wide applicability discussed in PSAR Section 3.1. During a DBA and assuming the unavailability of normal power, EFS does not operate, nor is it required to operate. The EFS does interface with containment and contain SC1 containment isolation for drain from containment floor drain to pressurized sump. As indicated in Table 6.2-2 “Process Line Isolation”, the EFS contains CIV to isolate drain line, which are evaluated in Section 6.2 of PSAR. Table 6.2-4 indicates these CIV are open during normal and hot shutdown but closed during post-accident and LOPP conditions. These design features ensure natural phenomena and accident conditions will not compromise required isolation capability. The NRC staff evaluated the EFS description provided in PSAR Section 9.3.3 and verified that the applicant has proposed compliance with the plant-wide requirements of GDC 2. Therefore, the NRC staff has confidence that an EFS design constructed in accordance with these criteria could meet the requirements of GDC 2 of Appendix A of 10 CFR Part 50. The NRC staff finds that the information provided by the applicant is sufficient and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35

PSAR Section 9.3.3.3 indicates EFS complies with GDC 4. The EFS is evaluated as moderate energy piping systems for potential candidates for postulated pipe crack, as indicated in PSAR Section 3.6.1. The NRC staff concludes that the EFS design will be appropriately evaluated for protection from dynamic effects of postulated pipe failures and therefore the NRC staff finds the design meets the intent of GDC 4.

The proposed design for Clinch River is a single-unit, single-module station, and the requirements of GDC 5 could be met.

GDC 60 requires control of release of radioactive material to the environment. Controlled collection and transfer to appropriate radioactive or nonradioactive treatment systems assures compliance with GDC 60 as described in SRP 9.3.3. To preclude inadvertent transfer of radioactive liquids to non-radioactive systems, PSAR Section 9.3.3.3 indicates the pipes carrying fluid to radioactive drains and clean drains are separate and independent. As described above, EFS contains sump types to collect and route fluid to the proper sump for treatment and disposal. Liquid waste collected in radioactive drains is treated as HCW. Therefore, the NRC staff finds the design meets the intent of GDC 60.

The flood analysis methodology and analysis results could potentially impact classification of design features such as drainage paths, sumps, pumps, or any combinations to mitigate the effects of flooding. Section 3.4.1.2 of PSAR specifies that the bounding results from the internal flooding analysis, including the limiting flood sources, bounding flooding levels, location of SSCs subject to flooding, and required protective and mitigation features, will be provided in the Final Safety Analysis Report (FSAR). The flood analysis was reviewed in section 3.4.1 of this report. This review was sufficient for the CP phase, because nothing requires a complete analysis in the PSAR. Use of EFS for flood mitigation will be further reviewed upon completion of flood analysis in Section 3.4.1 at the OL stage and classification of SSCs may need to be reevaluated.

9.3.3.4 Conclusion

TVA has described the equipment and floor drains in PSAR Section 9.3.3. The NRC staff finds that the content provided in the PSAR demonstrates that the preliminary design can meet GDC 5 and is consistent with GDC 2, 4, and 60. Based on its review and findings above, the NRC staff concludes that the preliminary design of the EFS is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.3.4 Chemical and Volume Control System

The proposed design does not include a chemical and volume control system and the PSAR, accordingly, does not describe any such system. The Reactor Water Cleanup (CUW) system, along with the Fuel Pool Cooling and Cleanup (FPCC) system, are described to manage water chemistry and volume control in the proposed plant.

9.3.5 Standby Liquid Control System

PSAR Section 9.3 discusses the standby liquid control system as it pertains to the CRN-1 design. The NRC staff notes that this information is not required to meet the applicable regulatory requirements for a CP under 10 CFR 50.33 through 50.35 and 10 CFR Part 50 (TN249), Appendix A. The NRC staff acknowledges that TVA included this information in its PSAR; however, the NRC staff has not evaluated this information as part of its safety review to support the issuance of the CP. Accordingly, the NRC staff does not take a position on the information presented in this section related to the standby liquid control system as it pertains to CRN-1.

9.3.6 Containment Inerting System

9.3.6.1 Introduction

The containment inerting system (CIS), as described in PSAR Section 9.3.6, is designed to establish and maintain an inert nitrogen atmosphere within the containment. The principal objective of the CIS is to prevent the development of a combustible atmosphere by maintaining an oxygen-deficient environment inside the containment. The CIS includes CIVs, which are evaluated in Section 6.2.4. The CIS supports the proposed design's containment combustible gas control, evaluated in Section 6.2.5.

The CIS is illustrated in PSAR Figure 9.3-5. It maintains an inert atmosphere in all operating modes, except during plant shutdown for refueling, equipment maintenance, and limited periods to permit inspection access at low reactor power. The CIS can de-inert the containment to allow safe personnel access without the need for a breathing apparatus. It also provides a pathway for fresh air supply into the containment during outages to maintain a breathable atmosphere for outage work.

9.3.6.2 Regulatory Evaluation

The following NRC regulations are relevant to this review:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.
- GDC 5, "Sharing of Structures, Systems, and Components," as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.
- GDC 41 requires that systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.
- GDC 42 requires that fission product control systems be designed to permit periodic inspections of important components.
- GDC 43 requires that fission product control systems be designed to permit periodic functional testing of important components.
- 10 CFR 50.44(c)(2) requires that all containments must have an inerted atmosphere or must limit hydrogen concentrations during and after an accident to less than 10 percent by volume, while maintaining containment structural integrity and appropriate accident mitigation features.

SRP 6.2.5, "Combustible Gas Control in Containment," Revision 3 (NRC 2021-TN8013) and SRP 6.5.3, "Fission Product Control Systems and Structures," Revision 3 (NRC 2021-TN8013) provide criteria to meet these regulatory requirements. The following regulatory guidance documents provide acceptable methods for compliance:

- RG 1.7, "Control of Combustible Gas Concentrations in Containment," Revision 3 describes acceptable methods for implementing 10 CFR 50.44 (NRC 2007-TN1977).

9.3.6.3 Technical Evaluation

GDC 5

GDC 5 requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units.
The CRN site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the CIS design can be in conformance with GDC 5.

GDC 41

The PSAR states that the CIS is designed to establish and maintain an inert nitrogen atmosphere within the containment in all operating modes, except during plant shutdown for refueling, equipment maintenance, and limited periods for inspection access at low reactor power. 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 PSAR also states that the FSAR will reflect full compliance with GDC 41 based on the final design of the containment and the combustible gas control and monitoring system, considering results from DBAs and severe accidents. The CIS includes the containment overpressure vent flow path to ensure containment design pressure is not exceeded during a severe accident. Sizing will be performed during detailed design.

The NRC staff, in its SE on NEDC-33911P-A, concluded that the proposal provides the capability to monitor and control hydrogen and oxygen concentrations in the containment atmosphere following postulated accidents, thereby maintaining containment integrity. The methodology described in NEDC-33911P-A was reviewed and approved by the NRC in a SE, dated March 12, 2021 (NRC 2021-TN13280). The CIS CIVs are evaluated and found acceptable in Section 6.2.4 of this report.

Based on the PSAR, the review and approval of NEDC-33911P that provided containment performance including the CIS, and the evaluation in Section 6.2.5, the NRC staff concludes that the proposed design meets GDC 41 requirements for the CIS, as it aligns with RG 1.7, SRP 6.2.5, and SRP 6.5.3.

GDC 42

The PSAR states that the CIS is designed to allow periodic inspection, operability testing, and leak rate testing. During plant operation, CIS valves, instrumentation, wiring, and other components located outside the containment can be visually inspected at any time.

Based on that, the NRC staff finds that the proposed design, as described in the PSAR, provides the capability for periodic testing and inspection of important components, consistent with SRP Section 6.5.3, and thus meets the requirements of GDC 42.

GDC 43

The PSAR states that the CIS is designed to permit periodic testing to demonstrate the ability of the CIS to meet design requirements. Preoperational tests document system reference characteristics (e.g., pressure differentials and flow rates), which serve as baselines for subsequent operational tests.

The NRC staff finds that the approach described in the PSAR supports periodic testing consistent with SRP 6.5.3 and thus meets the requirements of GDC 43.

10 CFR 50.44(c)(2)

The PSAR states that the proposed design features a dry, inerted containment that does not rely on combustible gas control to maintain safe shutdown and containment integrity following a DBA, meaning the hydrogen produced during the DBA is not expected to challenge containment integrity because the oxygen-deficient environment limits combustion. The proposed design uses Zircaloy-2 fuel cladding (Table 1.2-1, "Key Design Characteristics," of Enclosure 2 to the PSAR) which is one of the cladding materials used in light water reactor designs licensed as of October 16, 2003. Therefore, the proposed design has characteristics similar to the water-cooled reactor designs with (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to light water reactor designs licensed as of October 16, 2003. The NRC staff finds that this design supports control of combustible gas effects as required by 10 CFR 50.44.

PSAR section 6.2.5 also states that NEDC-33911P-A, which the NRC staff approved for referencing in licensing applications, addresses the functional capability of the considered combustible gas control systems. According to the applicant, the approach is consistent with 10 CFR 50.44(c)(2) and is acceptable for both normal operation and DBA conditions.

The NRC staff evaluated the proposed design containment combustible gas control in Section 6.2.5 and found it acceptable. Based on the finding earlier in this section about control of combustible gas effects and the applicant's acceptable reliance on NEDC-33911P-A, the NRC staff finds that the dry, inerted containment is consistent with RG 1.7 and SRP Section 6.2.5 and thus meets the requirements of 10 CFR 50.44(c)(2).

9.3.6.4 Conclusion

Based on its review, the NRC staff concludes that the applicant's preliminary design for the CIS provides reasonable assurance that the system will fulfill its intended functions once constructed, consistent with RG 1.7, SRP Sections 6.2.5 and 6.5.3. Therefore, the information provided by the applicant is sufficient and consistent with GDC 41, 42, 43. Further, based on the above discussion, the NRC staff concluded that the information is sufficient and can meet the requirements of 10 CFR 50.44(c)(2). Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.3.7 Hydrogen Water Chemistry Injection System

9.3.7.1 Introduction

The HWC injection system prevents and reduces the growth rates of intergranular stress corrosion cracking (IGSCC) in reactor vessel internals. This mitigation of IGSCC is achieved from the reduction of oxygen and other oxidizing species (oxidants) in the reactor coolant. This is accomplished by injecting hydrogen into the feedwater. The injected hydrogen suppresses the radiolytic formation of oxidants in the reactor core.

9.3.7.2 Regulatory Evaluation

The NRC staff finds the following regulations and codes and standards applicable to this review:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

Applicable GDC from 10 CFR Part 50, Appendix A:

- GDC 14 (Reactor Coolant Pressure Boundary) requires that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

Codes and Standards:

- EPRI BWRVIP-190, BWR Vessel and Internals Project, Volume 1: "BWR Water Chemistry Guidelines – Mandatory, Needed, and Good Practice Guidance" and Volume 2: "BWR Water Chemistry Guidelines – Technical Basis," consistent with guidance in SRP Section 5.4.8, "Reactor Water Cleanup System (BWR)," Revision 3, March 2007

9.3.7.3 Technical Evaluation

The NRC staff reviewed the applicant's description of the HWC injection system in PSAR Section 9.3.7. The HWC injection system does not perform a safety function, nor does it contain any safety class components. The HWC injection system is used, along with other measures, to reduce the likelihood of corrosion failures that would adversely affect plant availability. The PSAR describes components, instrumentation, inspection and testing of the system. The NRC staff finds that description of the HWC injection system acceptable because it demonstrates that the system is capable of reducing IGSCC in a manner consistent with the guidance in BWRVIP-190 as endorsed by SRP section 5.4.8. Therefore, the NRC staff finds that the proposal is consistent with GDC 14.

9.3.7.4 Conclusion

Based on its review, the NRC staff concludes that the applicant's preliminary design for the HWC Injection System demonstrates reasonable assurance that the system will fulfill its intended functions once constructed. The program will reduce IGSCC in a manner consistent with the guidance in BWRVIP-190 and will therefore be consistent with GDC 14. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.3.8 On-Line NobleChem System

9.3.8.1 Introduction

The purpose of the On-Line NobleChem (OLNC) system is to provide a means for the injection of a noble metal solution directly into the reactor coolant flow path to aid in the protection of reactor vessel internals from IGSCC, in combination with the addition of hydrogen by the HWC Injection System.

9.3.8.2 Regulatory Evaluation

The NRC staff finds the following regulations and codes and standards applicable to this review:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.
- GDC from 10 CFR Part 50, Appendix A:
  - GDC 14 (Reactor Coolant Pressure Boundary) requires that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

Codes and Standards:

- EPRI BWRVIP-190, BWR Vessel and Internals Project, Volume 1: "BWR Water Chemistry Guidelines – Mandatory, Needed, and Good Practice Guidance" and Volume 2: "BWR Water Chemistry Guidelines – Technical Basis", consistent with guidance in SRP Section 5.4.8, "Reactor Water Cleanup System (BWR)," Revision 3, March 2007.

9.3.8.3 Technical Evaluation

The NRC staff reviewed the applicant's description of the OLNC system in PSAR Section 9.3.8. The OLNC system does not perform a safety function, nor does it contain any safety class components. The PSAR describes components, instrumentation, inspection and testing of the system. The NRC staff finds that description of the OLNC system acceptable because it demonstrates that the system is capable of reducing IGSCC in a manner consistent with the guidance in BWRVIP-190 as endorsed by SRP section 5.4.8. Therefore, the NRC staff concludes that the proposed design is consistent with GDC 14.

9.3.8.4 Conclusion

Based on its review, the NRC staff concludes that the applicant's preliminary design for the OLNC system demonstrates reasonable assurance that the system will fulfill its intended functions once constructed. The program will reduce IGSCC in a manner consistent with the guidance in BWRVIP-190 and will therefore be consistent with GDC 14. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.3.9 Zinc Injection Passivation System

9.3.9.1 Introduction

The purpose of the zinc injection passivation system is to provide a means for injection of soluble zinc to reduce out of core shutdown dose rates. The zinc injection passivation system is used only during power operation in its normal operating mode.

9.3.9.2 Regulatory Evaluation

The NRC staff finds the following regulations and codes and standards applicable to this review:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.
- Applicable GDC from 10 CFR Part 50, Appendix A:
  - GDC 14 (Reactor Coolant Pressure Boundary) requires that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

Codes and Standards:

- EPRI BWRVIP-190, BWR Vessel and Internals Project, Volume 1: "BWR Water Chemistry Guidelines – Mandatory, Needed, and Good Practice Guidance" and Volume 2: "BWR Water Chemistry Guidelines – Technical Basis," consistent with guidance in SRP Section 5.4.8, "Reactor Water Cleanup System (BWR)," Revision 3, March 2007

9.3.9.3 Technical Evaluation

The NRC staff reviewed the applicant's description of the Zinc Injection Passivation System in PSAR Section 9.3.9. The Zinc Injection Passivation System does not perform a safety function, nor does it contain any safety class components. The PSAR describes components, instrumentation, inspection and testing of the system. The NRC staff finds that description of the zinc injection passivation system acceptable because it demonstrates that the system is capable of reducing IGSCC in a manner consistent with the guidance in BWRVIP-190 as endorsed by SRP section 5.4.8. Therefore, the NRC staff concludes that the proposed design is consistent with GDC 14.

9.3.9.4 Conclusion

Based on its review, the NRC staff concludes that the applicant's preliminary design for the zinc injection passivation system demonstrates reasonable assurance that the system will fulfill its intended functions once constructed. The program will reduce IGSCC in a manner consistent with the guidance in BWRVIP-190 and will therefore be consistent with GDC 14. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.3.10 Boron Injection System

9.3.10.1 Introduction

The BIS is a defense line DL4b function designed to introduce sufficient negative reactivity into the core to assure reactor shutdown from full power operating condition to a cold critical state.

9.3.10.2 Regulatory Evaluation

The following GDC from 10 CFR Part 50, Appendix A and applicant proposed PDCs apply to this system:

- GDC 1—Quality Standards and records
- GDC 2—Design basis for protection against natural phenomena
- GDC 4—Environmental and dynamic effect design bases
- GDC 5—Sharing of structures, systems and components
- PDC 26—Reactivity control system redundancy and capability

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."

- 10 CFR 50.35.

9.3.10.3 Technical Evaluation

As described in PSAR Section 9.3.10, the BIS system consists of a storage tank, a test tank, an injection pump, piping, valves, and instrumentation and controls necessary to prepare and inject a borated solution into the Reactor Coolant System as shown in CP Figure 9.3-6. The BIS is a defense line 4b system and mitigates design extension conditions (DECs) in the event that the normal means for reactivity control is unable to shutdown the reactor.

The BIS injects boron pentaborate solution into the ICS “C” loop condensate return line downstream of the two ICS return valves, as reviewed in Section 6.3. Injection into the ICS condensate return line provides a direct flow path to the reactor pressure vessel, ensuring that borated solution can be delivered to the core to achieve and maintain shutdown conditions during DECs when normal reactivity control is unavailable.

The BIS uses its own containment penetration and CIV upstream of the injection point. The system’s connection to Reactor Coolant System and containment penetration are reviewed in the respective Sections 5.2 and 6.2.4 of the SE.

The applicant identifies the BIS as the second reactivity control system which is independent and of a different design principle from the control rod system. Based on the system description and its credited function, the NRC staff finds that the BIS provides a reactivity control function that is independent and of a different design principle from the primary control rod system, consistent with PDC 26 as further evaluated in SE Sections 4.3 and 4.6.

Additional design details will be provided at the FSAR, including assurance that the BIS can reliably mitigate DECs and maintain the reactor subcritical under cold conditions.

The NRC staff finds that not providing detailed design information, including system performance characteristics, until the FSAR stage is acceptable because the PSAR provides sufficient information to establish the system’s basic design concept and safety function consistent with PDC 26. As such, more detailed information is not required at the CP stage. Demonstration of the system’s capability to reliably mitigate DECs and maintain the reactor subcritical under cold conditions will be evaluated at the OL stage when the design is finalized.

9.3.10.4 Conclusion

TVA has identified traits of the BIS in PSAR Section 9.3.10. The NRC staff reviewed the information provided for the BIS, including the system design, safety function, and its role in intended mitigation of design extension conditions.

As discussed above, the NRC staff finds that the preliminary design of the BIS is consistent with PDC 26 at the CP stage because it establishes a second, independent reactivity control system of a different design principle from the control rod system.

The NRC staff notes that additional design details will be provided in the FSAR, and compliance with all GDC will be confirmed as part of the OL review, should the applicant apply for an OL, when the design is finalized.

Because the BIS is a system credited for DECs and not design basis accidents, and because detailed design information is not required to be finalized at the CP stage, the NRC staff finds that the level of information provided in the PSAR is sufficient for evaluation under 10 CFR 50.35.

Based on the NRC staff's review and the findings above, the NRC staff concludes that the preliminary design of the BIS is sufficient and can meet requirements of PDC 26. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35. The GDC identified in Section 9.3.10.2 will be evaluated for full compliance at the OL stage when detailed design information becomes available.

9.4 Heating, Ventilation and Cooling Systems

9.4.1 Control Building Heating, Ventilation, and Cooling System

9.4.1.1 Introduction

This section identifies and describes the heating, ventilation, and cooling system (HVS) design that will be used to provide ventilation and maintain temperature, quality of air, and pressurization in areas of the CB. PSAR 9.4.1, "Control Building Heating, Ventilation, and Cooling System," discusses the plant design, which integrates shielding, ventilation, monitoring instrumentation with traffic control, access control, and health physics aspects to minimize radiation exposure to workers and to the public.

The CB receives heated, filtered, and conditioned air from one of two air handling units (AHUs), with one unit on standby for backup. Most air is recirculated with a small portion coming from outside. AHUs have adjustable speed drives to adjust airflow. High heat load rooms are additionally cooled by fan coil units, and non-vital areas use ductless mini-splits. The CB also has various exhaust fans: battery room fans operate on timers, while restroom and break room fans run continuously. Smoke exhaust fans are included, and electric duct heaters maintain optimal battery room temperatures. Tornado-rated dampers protect the HVAC system during high wind events.

The main control room (MCR) habitability is not a Safety Category 1 function, therefore the MCR envelope (MCRE) is not credited for mitigation of any design basis accidents. Instead, the CB HVS is analyzed for the effectiveness of preventing the spread of contamination into the MCRE. The MCRE consists of several areas in the CB including the MCR, emergency response facilities, access room, break room kitchenette, panel room, shift supervisor room, and toilet area. To protect MCR occupants from hazards like radiation, fire, or toxic gases, the system includes:

- Toxic gas filtration units (TGFUs) that activate automatically when toxic gases are detected at the CB air intake.
- Emergency filtration units (EFUs) that activate when high radiation is detected, isolating the MCRE and maintaining positive pressure.

Tornado-rated dampers protect the EFU intake/exhaust ducts by closing under extreme pressure conditions to prevent duct collapse.

In the event of a radiation alarm at the CB outside air intake, the operating AHU and TGFU shut down, and the corresponding MCRE EFU starts automatically. The MCRE isolation dampers close, and backup power from the standby diesel generator (SDG) supports the EFU fans. If toxic gas is detected, the AHU outside air intake closes, and the TGFU starts automatically, opening its dampers to supply filtered air for CB pressurization. SDG power supports the TGFU fans, but not the AHU heating coils. Positive pressure in the CB is maintained using air from the AHU, TGFU, or EFU. In case of fire, the CB HVS system isolates the MCRE and coordinates smoke exhaust with the fire protection system. During a LOPP or SBO, exhaust fans for restrooms, break room, and janitor's closet become inoperable. In an SBO, power is also lost to CB AHUs and heaters. If the MCR becomes uninhabitable, operators relocate to the secondary control room (SCR) in the RB.

The CB HVS is designed to permit periodic inspection and testing of major components, such as fans, motors, dampers, coils, filters, and ducts to verify their integrity, operability, and capability. CB HVS equipment and components are provided with proper access for initial and periodic inspection and maintenance activities.

9.4.1.2 Regulatory Evaluation

In accordance with 10 CFR 50.34(a)(3), "Contents of Applications; Technical Information", GDC 2, GDC 4, GDC 5, PDC 19, GDC 60, and 10 CFR 50.63 (TN249), the areas of the CB 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 regulations for this review:

- 10 CFR 50.34, "Contents of applications; technical information," including:
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.
- GDC 2, as CB HVS shall be capable of withstanding the effects of earthquakes.
- GDC 4, as CB HVS 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.
- GDC 5, as 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.
- The applicant is proposing to implement a design-specific PDC 19 in lieu of GDC 19. The NRC staff evaluates the acceptability of PDC 19 in Chapter 7 of this report.

- GDC 60, as CB HVS shall be capable to suitably control the release of gaseous radioactive effluents to the environment.
- 10 CFR 50.63, as necessary support systems shall provide sufficient capacity and capability to cope with an SBO event.

SRP Section 9.4.1, Revision 3, “Control Room Area Ventilation System,” provides acceptable criteria to meet the relevant regulatory requirements as well as interfaces with other sections.

- Regulatory (RG) 1.140 Revision 3, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” provides guidance and criteria acceptable for compliance with the NRC regulations regarding ventilation systems (NRC 2016-TN13277). Although this guidance is not applicable to CP applicants, the applicant has committed to following this RG as a means to address GDC 60. Since 50.34(a) requires information on the preliminary design of the facility, and RG 1.140 is applicable for the final design of the facility, it is also relevant in this context when cited by applicants.
- RG 1.78 Revision 2, “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 CR operators (NRC 2021-TN13248).
- RG 1.29 Revision 6, “Seismic Design Classification for Nuclear Power Plants,” provides acceptable methods for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the SSE (NRC 2021-TN12804).

9.4.1.3 Technical Evaluation

The NRC staff reviewed the Clinch River preliminary design provisions for the CB HVS in accordance with the guidance of SRP Section 9.4.1 and the requirements listed above. The results of the NRC staff’s review are provided below.

The NRC staff based its review of CB HVS compliance with GDC 2 requirements on consistent with RG 1.29, Regulatory Position C.2. Based on its review of the PSAR, the NRC staff understands that the MCRE is not credited for mitigation of any DBAs and the CB HVS pressurization and isolation functions are Safety Category 3 and non-seismic. The NRC staff notes that the CB including the MCRE is a Seismic Category II structure, and a seismic event of sufficient magnitude may result in the MCR becoming functionally impaired.

RG 1.29 C.2 indicates, “Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level or could result in incapacitating injury to occupants of the CR, should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.”

The applicant indicated in PSAR Section 9.4.1.3, that during a significant seismic event, the SCR is the assured shutdown location due to the seismic rating of the CB being a Seismic Category I structure. The NRC staff reviewed the associated egress route between the MCR and SCR and confirmed that human occupants of those areas are protected and not caused incapacitating physical harm as a result of any design basis natural phenomena event.

The NRC staff finds the proposed design is consistent with the guidance of RG 1.29, therefore, concludes that the proposed design is consistent with the requirements of GDC 2 as it relates to CB HVS.

The CB HVS is designed to maintain a suitable ambient temperature, quality air, and pressurization for personnel and equipment in the MCR and other areas of the CB during normal operation and off-normal operation. The CB HVS has radiation monitors and gas/smoke detectors located in the outside air intake and downstream ductwork, which allow the plant protection system or plant control system to isolate the MCRE and the outside air intake as needed in the event of high radiation levels from DBAs, release of radioactive material, fires, or explosive toxic gases. Tornado dampers are provided on CB EFU air intake/exhaust openings and are designed to withstand high wind events. The applicant indicated that testing and ratings of tornado dampers are in accordance with Air Movement and Control Association (AMCA) Standard 500-D, "Laboratory Methods of Testing Dampers for Rating."

The NRC staff confirmed that the CB HVS SSCs, including MCRE isolation dampers, are compatible with the environmental conditions during normal and off-normal operation, and the dynamic effects present during the event scenarios where these functions are credited, including the effects of tornados, and therefore concludes that the proposed design is consistent with the requirements of GDC 4 as it relates to CB HVS.

The CRN-1 is a single unit, single-module facility, and the CB HVS does not share any SSCs with other nuclear power plants. Therefore, the NRC staff concludes that the requirements of 10 CFR Part 50, Appendix A, GDC 5 can be met by the design.

PDC 19, which the NRC staff evaluated the acceptability of in Chapter 7 of this report, requires that nuclear plants operate safely under normal conditions, AOOs, and DBAs, while ensuring adequate radiation protection, human factors, and CR habitability (e.g., HVS and lighting). The proposed design meets these requirements by limiting radiation exposure in the MCR or SCR to below 5 rem TEDE during DBAs.

The applicant used the guidance of RG 1.78 to assess meeting the MCR occupancy protection requirements for postulated hazardous chemical releases. For these credible postulated chemical releases, design features of the CB HVS offer protection for operators in the MRC, as discussed in the rest of this paragraph. In the event of toxic gas detection, the AHUs outside air intake closes, and the TGFU automatically activates. Its motorized dampers open to deliver filtered air, maintaining pressurization in the CB. The SDG supplies power to the TGFU fans, but not to the AHUs heating coils. If a radiation alarm is triggered at the CB outside air intake, the active AHU and TGFU automatically shut down, while the corresponding MCRE EFU starts. Simultaneously, the MCRE isolation dampers close, and the SDG provides backup power to the EFU fans. In the case of a fire, the CB HVS system isolates the MCRE and works with the fire protection system to manage smoke exhaust.

The design also includes remote safe shutdown capabilities outside the MCR, consistent with USNRC guidance (e.g., SECY-94-084 and RG 1.189, as discussed in the rest of this paragraph). 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 RB. Accordingly, the NRC staff finds that this meets the remote shutdown portion of PDC 19 by providing means for operators to maintain the reactor in a safe condition in the event of a CR evacuation.

For the reasons discussed above, the NRC staff finds that the CB HVS design ensures safe operation and shutdown under all conditions, including emergencies requiring MCR evacuation, and that the guidance of RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release,” has been followed. Therefore, the NRC staff concludes that the proposed design complies with the objectives of PDC 19.

The CB does not contain any components of the nuclear steam supply system or other equipment that could serve as a source of radioactive material. As a result, there are no anticipated sources of radioactive particulates or gases within the CB. Nevertheless, appropriate controls are in place to manage the potential release of gaseous radioactive effluents to the environment.

Since the proposed design does not utilize an engineered safety feature (ESF) ventilation system, the NRC staff considers that RG 1.140 is applicable to the CB HVS because it describes the supply air atmospheric cleanup function. According to RG 1.140, standards acceptable to the NRC staff for the design and testing of the system include ASME AG-1, “Code on Nuclear Air and Gas Treatment.”

The NRC staff finds that the CB HVS design accurately considers the guidance for RG 1.140 because the design and testing of the system considers the use of the ASME AG-1 Code. Therefore, the NRC staff concludes that the proposed design description is adequate to address GDC 60, as it relates to the design and testing of the CB HVS. The NRC staff will confirm the final design’s compliance with GDC 60 at the OL stage because this information is not required to support issuance of the CP.

In the event of a SBO, power is lost to the CB AHUs, as well as electric duct heaters and cabinet heaters. The restroom, break room, and janitors closet exhaust fans become inoperable. The CR operators relocate to the SCR in the RB if the MCR becomes uninhabitable.

The NRC staff notes that this information is not required to meet the applicable regulatory requirements for a CP under 10 CFR 50.33 through 50.35 and 10 CFR Part 50, Appendix A. The NRC staff acknowledges that TVA included this information in its PSAR; however, the NRC staff has not evaluated this information as part of its safety review to support the issuance of the CP. Accordingly, the NRC staff does not take a position on the information presented in this section related to compliance with the requirements of 10 CFR 50.63

9.4.1.4 Conclusion

The NRC staff reviewed the applicant’s preliminary CB HVS design for Clinch River and concludes it is consistent with the NRC regulations as set forth in Appendix A, GDC 2, 4, 5, and 60, as well as the custom PDC 19 for a controlled environment for the comfort and safety of CR personnel and assures the operability of CR components during norming operating, anticipated operational transient, and DBA conditions. The NRC staff concludes this based on the applicant addressing the applicable GDCs listed in SRP Section 9.4.1 and because the applicant used acceptable methods in accordance with RG 1.140 for ventilation systems, and RG 1.78 for hazardous chemical releases. Further, the NRC staff concludes, based on the above discussion, that the applicant meets the requirements of 10 CFR 50.34(a)(3). The NRC staff finds that the information provided by the applicant is sufficient to address the regulatory requirements identified in this section and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.4.2 Fuel Pool Area Ventilation

The proposed design does not have a dedicated ventilation system for the fuel pool area. The fuel pool is located in the RB and is ventilated by the RB HVS. As discussed in Section 9.4.6 of this safety evaluation report, the RB HVS provides an exhaust function above the pool and in response to a fuel handling accident, the fuel pool area is isolated by the automatic closure of supply and exhaust dampers.

9.4.3 Radioactive Waste Building Heating, Ventilation, and Cooling System Radioactive Waste Building Heating, Ventilation, and Cooling System

9.4.3.1 Introduction

This section identifies and describes the NRC's staff review of the heating, ventilation, and cooling system (HVS) design that will be used to provide ventilation and maintain temperature, quality of air, and pressurization in areas of the RWB. PSAR 9.4.3, "Radwaste Building Heating, Ventilation, and Cooling System," discusses the plant design to support personnel access and equipment functions by maintaining a suitable operating environment in the RWB. Plant personnel can access the Chem Laboratory and dress out areas within the RWB.

The RWB ventilation system uses a push-pull setup with supply and exhaust AHUs. Clean, filtered outdoor air is supplied to the lobby and lab through ductwork, pressurizing these clean areas. Meanwhile, exhaust AHUs draw air from potentially contaminated spaces like tank and pump rooms, creating negative pressure and ensuring airflow moves from clean to dirty zones. Exhaust air is released to the atmosphere via the plant vent stack (PVS).

To support temperature control:

- Fan coil units (FCUs) provide additional cooling in the lobby, lab, and upper main area, using chilled water.
- The lab has a cabinet heater, and the lobby includes electric unit heaters.
- The chemical lab fume hood exhaust is routed to the RWB exhaust AHUs.

In the event of a fire, the system also functions to remove smoke during recovery.

The RWB HVS is designed to permit periodic inspection and testing of major components, such as fans, motors, dampers, coils, filters, ducts, and piping to verify their integrity and capability. RWB HVS equipment and components are provided with proper access for inspection and maintenance activities, and instruments are calibrated during testing.

9.4.3.2 Regulatory Evaluation

In accordance with 10 CFR 50.34(a)(3) (TN249), "Contents of Applications; Technical Information", GDC 2, 5, and 60, the areas of the RWB must be designed to permit operator access and control the concentration of airborne radioactive material during normal operation and after postulated accidents.

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

- 10 CFR 50.34, "Contents of applications; technical information," including:

– 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

– 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

– 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."

- 10 CFR 50.35.
- GDC 2, as the RWB HVS shall be capable of withstanding the effects of earthquakes.
- GDC 5, as 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.
- GDC 60, as the RWB HVS shall be capable of suitably controlling the release of gaseous radioactive effluents to the environment.

SRP Section 9.4.3 Revision 3, "Auxiliary and Radwaste Area Ventilation System," provides the criteria to meet the relevant regulatory requirements as well as interfaces with other sections.

- Regulatory (RG) 1.140 Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," provides guidance and criteria acceptable for compliance with the NRC regulations regarding ventilation systems.
- RG 1.29 Revision 6, "Seismic Design Classification for Nuclear Power Plants," provides acceptable methods for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the SSE (NRC 2021-TN12804).

9.4.3.3 Technical Evaluation

The NRC staff reviewed the Clinch River preliminary design provisions for the RWB HVS in accordance with the guidance of SRP Section 9.4.3 and the requirements listed above. The results of the NRC staff's review are provided below.
The NRC staff based its review of RWB HVS compliance with GDC 2 requirements consistent with RG 1.29, Regulatory Position C.2. Based on its review of the PSAR, the NRC staff understands that the RWB HVS is not credited for mitigation of any DBAs and the RWB HVS pressurization functions are Safety Category N and Non-Seismic. The NRC staff notes that the RWB is categorized as Seismic Category RW-IIa, per RG 1.143 (NRC 2001-TN1134), and is designed for 1/2 SSE. Additionally, the RWB is adjacent to and can interact with the RB during seismic or extreme wind events.

RG 1.29 C.2 indicates "Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level, or could result in incapacitating injury to occupants of the CR, should be designed and constructed so that the SSE would not cause such failure.

Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility."

The applicant stated in PSAR Section 3.7, that the seismic interaction evaluation of the RWB ensures that the structural integrity and safety functions of the RB Seismic Category I SSCs are not compromised. Therefore, the NRC staff finds that the guidance of RG 1.29 has been satisfied and therefore concludes that the proposed design is consistent with the requirements of GDC 2 as it relates to RWB HVS. A more detailed evaluation of seismic analysis of this system is contained in Chapter 3 of this SER.

The CRN-1 is a single unit, single-module facility, and the RWB HVS does not share any SSCs with other nuclear power plants. Therefore, the NRC staff concludes that the requirements of 10 CFR Part 50, Appendix A, GDC 5 can be met by the design.

Air from the RWB HVS system is exhausted through active AHUs and released to the atmosphere via the PVS. The effluent is monitored for radiation by the plant radiation monitor with one radiation monitor installed upstream of the exhaust AHUs. The system follows RG 1.143, which aligns with 10 CFR Part 20 (TN283) requirements for gaseous radioactive waste management. These measures ensure that radioactive releases in gaseous effluents, including during AOOs, are properly controlled and monitored.

Since the proposed design does not utilize an ESF ventilation system, the NRC staff considers that RG 1.140 is applicable to RWB HVS because it describes the air atmospheric cleanup function. Further, since this guidance is applicable at the OL phase and the CP phase involves preliminary information for the OL phase, the NRC staff considered it acceptable for use at this stage. According to RG 1.140, standards acceptable to the NRC staff for the design and testing of the system include ASME AG-1, "Code on Nuclear Air and Gas Treatment."

The NRC staff finds that the RWB HVS design follows the guidance for RG 1.140 because the design and testing of the system use the ASME AG-1 Code. Therefore, the NRC staff concludes that the proposed design is consistent with GDC 60, as it relates to the design and testing of the RWB HVS.

9.4.3.4 Conclusion

The NRC staff review concludes that the applicant's preliminary RWB HVS design for Clinch River to be consistent with the NRC regulations as set forth in Appendix A, GDC 2, 5, and 60, for a controlled environment for the access of plant personnel and assures the control of airborne radioactive material in the RWB during norming operating, anticipated operational transient, and DBA conditions based on the applicant addressing the applicable GDCs listed in SRP Section 9.4.3 and because the applicant used acceptable methods in accordance with RG 1.140 for ventilation systems. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.4.4 Turbine Building Heating, Ventilation, and Cooling System

9.4.4.1 Introduction

This section identifies and describes the heating, ventilation, and cooling system (HVS) design that will be used to provide ventilation and maintain temperature, quality of air, and pressurization in areas of the TB. PSAR Section 9.4.4, “Turbine Building Heating, Ventilation, and Cooling System,” discusses the plant design, which integrates shielding, ventilation, monitoring instrumentation with traffic control, access control, and health physics aspects to minimize radiation exposure to workers and to the public.

The TB receives outside air through supply AHUs equipped with electric heating coils and prefilters to maintain air temperature and cleanliness. Air is distributed via ductwork to areas both inside and outside the bioshield. The building’s metal grating floors allow warm air from lower levels to rise naturally.

Exhaust AHUs remove air through roof penetrations or ductwork, with airflow designed to move from clean zones (outside the bioshield) to potentially contaminated zones (inside the bioshield). All TB AHUs include ASDs to modulate airflow based on operational and ambient conditions.

The HVS air supply near the main steam piping is specifically designed to minimize turbulence, ensuring that a reliable high-temperature signal can be obtained in the turbine area. Each TB Switchgear room is equipped with an FCU for cooling.

The TB Battery Room is ventilated and climate-controlled using two mini-split HVS systems. One unit is backed by SDG Train A, and the other by SDG Train B, providing redundancy. One of these systems also includes electric unit heaters to maintain the room within its design temperature range. Both mini-split units introduce a small amount of filtered, conditioned outside air, maintaining a slightly positive pressure in the room. To manage hydrogen off-gassing from the batteries, exhaust ventilation is provided and operates on a timer, based on the manufacturer’s specified off-gassing rate. Exhausted air from potentially contaminated areas within the TB, RB, and RWB is directed to the continuous exhaust air plenum. This plenum functions as a large mixing chamber, where exhaust air is blended and diluted. During normal operation, the PVS fans draw air from the plenum and discharge it to the PVS. These fans are equipped with ASDs, allowing them to:

- Maintain the plenum at a negative pressure relative to the outside environment.
- Adapt to varying airflow rates, which depend on the operation of AHUs in each building.

The TB HVS is designed to permit periodic inspection and testing of major components, such as fans, motors, dampers, coils, filters, ducts, piping and valves to verify their integrity and capability. TB HVS equipment and components are provided with proper access for inspection and maintenance activities, and instruments are calibrated during testing.

9.4.4.2 Regulatory Evaluation

In accordance with 10 CFR 50.34(a)(3) (TN249), “Contents of Applications; Technical Information”, GDC 2, 5, and 60, the areas of the turbine building must be designed to permit personnel access and control the concentration of airborne radioactive material in the TB during normal operation, AOOs, and following any accident that results in a release of radioactive material.

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

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, “Contents of applications; technical information,” including:

- 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
- 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
- 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."

- 10 CFR 50.35.
- GDC 2, the NRC requires, in part, that the turbine building HVS shall be capable of withstanding the effects of earthquakes.
- 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 60, the NRC requires, in part, that the turbine building HVS shall be capable of suitably controlling the release of gaseous radioactive effluents to the environment.

SRP Section 9.4.4 Revision 3, "Turbine Area Ventilation System," provides the criteria to meet the relevant regulatory requirements as well as interfaces with other sections.

- Regulatory (RG) 1.140 Revision 3, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," provides guidance and criteria acceptable for compliance with the NRC regulations regarding ventilation systems.
- RG 1.29 Revision 6, "Seismic Design Classification for Nuclear Power Plants," provides acceptable methods for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the SSE (NRC 2021-TN12804).

9.4.4.3 Technical Evaluation

The NRC staff reviewed the Clinch River preliminary design provisions for the TB HVS in accordance with the guidance of SRP Section 9.4.4 and the requirements listed above. The results of the NRC staff's review are provided below.

The NRC staff based its review of TB HVS compliance with GDC 2 requirements consistent with RG 1.29, Regulatory Position C.2. Based on its review of the PSAR, the NRC staff understands that the TB HVS is not credited for mitigation of any DBAs and the TB HVS pressurization functions are Safety Category N and non-seismic. The NRC staff notes that the TB is categorized as Seismic Category II. Additionally, the TB has the potential to interact with the RB during seismic or extreme wind events due to the proximity to the RB.

RG 1.29 C.2 indicates "Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level, or could result in incapacitating injury to occupants of the CR, should be designed and constructed so that the SSE would not cause such failure.

Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility."

The applicant indicated in PSAR Section 3.7, that the seismic interaction evaluation of the TB ensures that the structural integrity and safety functions of the RB Seismic Category I SSCs are not compromised. Therefore, the NRC staff finds that the guidance of RG 1.29 has been satisfied and therefore concludes that the proposed design is consistent with the requirements of GDC 2 as it relates to TB HVS. A more detailed evaluation of seismic analysis of this system is contained in Chapter 3 of this report.

The CRN-1 is a single-unit, single-module facility, and the TB HVS does not share any SSCs with other nuclear power plants. Therefore, the NRC staff concludes that the requirements of 10 CFR Part 50, Appendix A, GDC 5 can be met by the design.

Air from the TB HVS system is exhausted through AHUs and released to the atmosphere via the PVS. The effluent is monitored for radiation by the plant radiation monitor with one radiation monitor installed upstream of each exhaust AHU set (total 2 monitors). The system follows RG 1.143, which aligns with 10 CFR Part 20 requirements for gaseous radioactive waste management. These measures ensure that radioactive releases in gaseous effluents, including during AOOs, are properly controlled and monitored.

Since the proposed design does not utilize an ESF ventilation system, the NRC staff considers that RG 1.140 is applicable to the TB HVS because it describes the air atmospheric cleanup function. Further, since this guidance is applicable at the OL phase and the CP phase involves preliminary information for the OL phase, the NRC staff considered it acceptable for use at this stage. According to RG 1.140, standards acceptable to the NRC staff for the design and testing of the system include ASME AG-1, "Code on Nuclear Air and Gas Treatment."

The NRC staff finds that the TB HVS design follows the guidance for RG 1.140 because the design and testing of the system use the ASME AG-1 Code. Therefore, the NRC staff concludes that the proposed design is consistent with GDC 60, as it relates to the design and testing of the TB HVS.

9.4.4.4 Conclusion

The NRC staff review concludes that the applicant's preliminary RWB HVS design for Clinch River is consistent with the NRC regulations as set forth in Appendix A, GDC 2, 5, and 60, for a controlled environment for the access of plant personnel and assures the control of airborne radioactive material in the TB during norming operating, anticipated operational transient, and DBA conditions based on the applicant addressing the applicable GDCs listed in SRP Section 9.4.4 and because the applicant used acceptable methods in accordance with RG 1.140 for ventilation systems. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.4.5 Engineered Safety Feature Ventilation System

The applicant states that this section is not applicable to the proposed design since engineered safety features for ventilation systems to mitigate the offsite consequences of a DBA are not utilized. Based on the passive design, the NRC staff agree that the ESF for ventilations systems are not applicable.

9.4.6 Reactor Building Heating, Ventilation, and Cooling System

9.4.6.1 Introduction

This section identifies and describes the heating, ventilation, and cooling system (HVS) design that will be used to provide ventilation and maintain temperature, quality of air, and pressurization in areas of the RB, which includes the fuel pool and the SCR. PSAR 9.4.6, "Reactor Building Heating, Ventilation, and Cooling System," discusses the plant design, which integrates shielding, ventilation, monitoring instrumentation with traffic control, access control, and health physics aspects to minimize radiation exposure to workers and to the public.

The lower levels of the RB receive heated, filtered, and conditioned once-through air from one of two AHUs, with the second unit in standby mode for automatic startup upon failure of the primary. Adjustable speed drives (ASDs) allow airflow modulation to maintain proper static pressure. Electric duct heaters in the supply ducts to the battery rooms and SCR maintain temperatures near the battery manufacturer's recommended range, with chilled water AHUs providing cooling. To manage hydrogen off-gassing from batteries, timed exhaust ventilation is used based on manufacturer-specified off-gassing rates. Additional cooling is provided by fan coil units (FCUs) in rooms with higher cooling demands, controlled by room thermostats. The RB HVS system is specifically designed to keep the electrical distribution system battery rooms within the optimal temperature range during normal operation.

The upper levels of the RB, including the fuel handling area and operating deck, receive heated, filtered, and conditioned once-through air from one of two AHUs, with the second unit in standby for automatic startup if the primary fails. ASDs allow airflow adjustment, particularly to the refueling pool. In the event of a fuel handling accident, a radiation monitor alarm triggers automatic closure of supply and exhaust isolation dampers and shutdown of the upper-level AHUs. Air from both the upper and lower RB levels, including the battery rooms, is exhausted through one of two exhaust AHUs, which discharge to the plant vent stack (PVS).

When needed, the RB exhaust system receives air and inerting gas from the CIS for venting through the PVS. Additionally, the lower-level supply AHUs provide outside air to support containment de-inerting and ensure a habitable environment for personnel during refueling and maintenance activities.

During a LOPP, the SDG supplies power to one RB exhaust AHU fan and one supply fan each for the upper and lower RB levels, maintaining basic ventilation. However, backup power is not available for the supply AHU heating coils or the battery room electric duct heaters. In a SBO, power is lost to all RB exhaust and supply AHUs and battery room heaters. Cooling is not required for at least the first 72 hours of the SBO event.

The RB HVS is designed to permit periodic inspection and testing of major components, such as fans, motors, dampers, coils, filters, ducts, piping, and valves to verify their integrity and capability. RB HVS equipment and components are provided with proper access for initial and periodic inspection and maintenance activities.

9.4.6.2 Regulatory Evaluation

In accordance with 10 CFR 50.34(a)(3), "Contents of Applications; Technical Information", GDC 2, 4, 5, 60, 61, PDC 19, and 10 CFR 50.63 (TN249), the areas of the RB 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:

- 10 CFR 50.34, “Contents of applications; technical information,” including:

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include “Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.”

- 10 CFR 50.35.
- GDC 2, as the RB HVS shall be capable of withstanding the effects of earthquakes.
- GDC 4, as the RB HVS 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.
- GDC 5, as 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.
- The applicant is proposing to implement a design-specific PDC 19 in lieu of GDC 19. The NRC staff evaluates the acceptability of PDC 19 in Chapter 7 of this report.
- GDC 60, as the RB HVS shall be capable to suitably control the release of gaseous radioactive effluents to the environment.
- 10 CFR 50.63, as necessary support systems shall provide sufficient capacity and capability to cope with an SBO event.

SRP Section 9.4.1 Revision 3, “Control Room Area Ventilation System,” and SRP 9.4.2, “Spent Fuel Pool Area Ventilation System,” provides the criteria to meet the relevant regulatory requirements as well as interfaces with other sections.

- RG 1.140 Revision 3, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” provides guidance and criteria acceptable for compliance with the NRC regulations regarding ventilation systems (NRC 2016-TN13277).
- RG 1.78 Revision 2, “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 CR operators (NRC 2021-TN13248).
- RG 1.29 Revision 6, “Seismic Design Classification for Nuclear Power Plants,” provides acceptable methods for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the SSE (NRC 2021-TN12804).

9.4.6.3 Technical Evaluation

The NRC staff reviewed the Clinch River preliminary design provisions for the RB HVS in accordance with the guidance of SRP Sections 9.4.1 and 9.4.2, and the requirements listed above. The results of the NRC staff's review are provided below.

The NRC staff based its review of RB HVS compliance with GDC 2 requirements on adherence to RG 1.29, Regulatory Position C.2. Based on its review of the PSAR, the NRC staff understands that the secondary control room envelope (SCRE) is not credited for mitigation of any DBAs and the SCR Emergency HVS pressurization and isolation functions are Safety Category 2 and non-seismic. Additionally, the NRC staff notes that the refueling floor isolation dampers and battery room exhaust fans are Safety Category 3 and non-Seismic.

RG 1.29 C.2 indicates "Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level or could result in incapacitating injury to occupants of the CR, should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility."

The licensee indicated that during a significant seismic event, the SCR is the assured shutdown location due to the seismic rating of the RB being a Seismic Category I structure. The NRC staff reviewed the associated egress route between the MCR and SCR and confirmed that human occupants of those areas are protected and not caused incapacitating physical harm as a result of any design basis natural phenomena event.

The NRC staff finds that the guidance of RG 1.29 has been satisfied and therefore concludes that the proposed design is consistent with the requirements of GDC 2 as it relates to RB HVS.

The RB HVS is designed to maintain a suitable ambient temperature, quality air, and pressurization for personnel and equipment in the SCR, the fuel pool area, and other areas of the RB during normal operation and off-normal events. The SCR HVS in the RB has radiation monitors and gas/smoke detectors located in the outside air intake and downstream ductwork, which allow the plant protection system or plant control system to isolate the SCRE and the outside air intake as needed in the event of high radiation levels from DBAs, release of radioactive material, fires, or explosive toxic gases. Additionally, the fuel pool area has radiation monitors to isolate the RB upper level and the outside air exhaust as needed in the event of high radiation levels from a fuel handling accident. Tornado dampers are provided on all RB HVS air intake/exhaust openings and are designed to withstand high wind events. The applicant indicated that testing and ratings of tornado dampers are in accordance with Air Movement and Control Association (AMCA) Standard 500-D, "Laboratory Methods of Testing Dampers for Rating."

The NRC staff confirmed that the RB HVS SSCs, including SCRE isolation dampers and refueling floor isolation dampers, are compatible with the environmental conditions during normal and off-normal operation, and the dynamic effects present during the event scenarios where these functions are credited, including the effects of tornados, and therefore concludes that the proposed design is consistent with the requirements of GDC 4 as it relates to RB HVS.

The CRN-1 is a single unit, single-module facility and the CB HVS does not share any SSCs with other nuclear power plants. Therefore, the NRC staff concludes that the requirements of 10 CFR Part 50, Appendix A, GDC 5 can be met by the design.

PDC 19, which the NRC staff evaluated the acceptability of in Chapter 7 of this report, requires that nuclear plants operate safely under normal conditions, AOOs, and DBAs, while ensuring adequate radiation protection, human factors, and CR habitability (e.g., HVS and lighting). The —proposed design meets these requirements by limiting radiation exposure in the MCR or SCR to below 5 rem TEDE during DBAs.

The SCR normally receives filtered and conditioned air from the RB lower-level AHU. It is equipped with two trains of EFUs and pressurization fans, which automatically activate upon detection of radiation, toxic gas, smoke, or loss of power to both normal supply AHUs. These systems are powered by the emergency power system and backed by battery power. The pressurization fan draws air through dedicated ductwork in the hallway and truck bay exterior walls. During an SBO, the SCR remains passively cooled for 72 hours via the thermal mass of the concrete structure and heat transfer to the surrounding ground. When power and CWE are available, two FCUs provide supplemental cooling. Additionally, electric duct heaters in the emergency supply ducting can heat incoming outside air if needed.

The MCR in the CB is the primary location for plant operations and habitability, as required by PDC 19. The SCR in the RB serves as a backup for safe plant shutdown if the MCR is evacuated. This is consistent with NRC guidance (e.g., SECY-94-084 and RG 1.189, as discussed in the rest of this paragraph). 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 RB. Accordingly, the NRC staff finds that this is consistent with the remote shutdown portion of PDC 19 by providing means for operators to maintain the reactor in a safe condition in the event of a CR evacuation.

The plant is designed so that no DBA can impact both CRs simultaneously, ensuring at least one remains habitable. While the hazardous chemical screening for CRN-1, based on RG 1.78, does not assume a simultaneous event requiring relocation, the SCR is still capable of supporting safe operations under accident conditions.

For the reasons discussed above, the NRC staff finds that the RB HVS design ensures safe operation and shutdown under all conditions, and that the guidance of RG 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release,” has been followed. Therefore, the NRC staff concludes that the proposed design complies with the requirements of PDC 19.

The RB HVS is designed with area and process radiation monitors within the RB that continuously measure radiation levels. If radiation exceeds a defined setpoint, the system automatically isolates the building. This design approach ensures that the release of radioactive materials in gaseous effluents remains within the limits specified by 10 CFR Part 20 (TN283) during normal operations and AOOs and therefore the NRC staff concludes that the proposed design is consistent with GDC 60, as it relates to the design and testing of the RB HVS.

The NRC staff reviewed the RB HVS using RG 1.13 (NRC 2007-TN13096), Regulatory Position C.4, which states that a controlled-leakage building should enclose the fuel to limit the potential release of radioactive iodine and other radioactive materials. The fuel pool is located in the RB, which is a controlled-leakage building.

The RB HVS provides containment and confinement to minimize the release of airborne radioactivity from the fuel pool during both normal operations and postulated accident conditions. The section of the HVS serving the refueling floor can be automatically isolated in response to signals from monitoring instrumentation. In the event of a fuel handling accident, a radiation monitor alarm triggers the closure of RB isolation dampers and shutdown of the upper-level AHUs, effectively containing any potential release of radioactive gases.

The NRC staff finds that the above design is consistent with guidance of RG 1.13, Regulatory Position C.4 and therefore concludes that the proposed design is consistent with the requirements of GDC 61 as it relates to RB HVS.

In the event of a SBO, power is lost to the RB exhaust AHUs, the upper and lower-level supply AHUs, as well as electric duct heaters for the Battery Rooms. No cooling is needed for at least the first 72 hours of the SBO event.

The SCR, digital control and instrumentation system (DCIS), and electrical equipment rooms are passively cooled for the full 72-hour battery coping duration by the thermal mass of the surrounding concrete structure and adjacent ground. This ensures the SCR remains habitable and that temperatures in the DCIS and electrical rooms stay within the qualified limits for their equipment. The emergency power system, supported by battery backup, maintains essential accident monitoring and SCR habitability functions during this period.

The NRC staff notes that this information is not required to meet the applicable regulatory requirements for a CP under 10 CFR 50.33 through 50.35 and 10 CFR Part 50, Appendix A. The NRC staff acknowledges that TVA included this information in its PSAR; however, the NRC staff has not evaluated this information as part of its safety review to support the issuance of the CP. Accordingly, the NRC staff does not take a position on the information presented in this section related to compliance with the requirements of 10 CFR 50.63.

9.4.6.4 Conclusion

The NRC staff review concludes that the applicant's preliminary RB HVS design for Clinch River is consistent with the NRC regulations as set forth in 10 CFR Part 50Appendix A, GDC 2, 4, 5, 60, and 61, as well as the custom PDC 19 for a controlled environment for the comfort and safety of RB personnel and assures the operability of RB components during norming operating, anticipated operational transient, and DBA conditions. This is the NRC staff's conclusion based on the applicant addressing the applicable GDCs listed in SRPs Sections 9.4.1 and 9.4.2 and because the applicant used acceptable methods in accordance with RG 1.78 for hazardous chemical releases, RG 1.13, RG 1.189, SECY-94-084, and RG 1.29 for earthquakes. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.4.7 Containment Cooling System

9.4.7.1 Introduction

This section identifies and describes the cooling system design that will be used to maintain the temperature for plant personnel and equipment inside the containment. PSAR Section 9.4.7, "Containment Cooling System (CCS)," discusses the plant design, which uses AHUs with cooling coils that reject heat to the CWE.

The CCS provides SC3 functions to maintain the containment bulk average temperature within the environmental qualification limits of the related equipment inside the containment during shutdowns and refueling activities, and to assist with containment cooldown following a LOPP and during periods of hot and cold shutdowns. SCN functions include maintaining the temperature for plant personnel entering containment during shutdowns and refueling activities.

9.4.7.2 Regulatory Evaluation

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

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

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

- RG 1.29 Revision 6, "Seismic Design Classification for Nuclear Power Plants," provides acceptable methods for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the SSE (NRC 2021-TN12804).

9.4.7.3 Technical Evaluation

The NRC staff reviewed the Clinch River preliminary design provisions for the CCS in accordance with the requirements listed above. The results of the NRC staff's review are provided below.

The CCS is a closed-loop air or nitrogen recirculation system designed to maintain temperature within the containment. It consists of two fully redundant trains, each with two 50 percent capacity AHUs. These AHUs include fans, cooling coils, and filters, and are entirely located within the containment.

Each CCS train is cooled by a corresponding CWE train, allowing continued operation even if one CCS or CWE train fails. Condensate from the cooling coils is drained to the EFS.

Ductwork distributes cooled air or nitrogen throughout the containment and CRD area. Return ducts from upper areas help reduce thermal stratification and recirculate air back to the AHUs. The system does not introduce external air or nitrogen; instead, the air is purged and inerted with nitrogen via the CIS before startup and during operation.

One CCS train operates as the lead, initiated remotely from the MCR, while the other remains in standby. The CWE temperature control valve regulates chilled water flow to maintain the desired supply air temperature. The standby train starts automatically if the lead train trips or if containment temperature exceeds a setpoint. Each AHU can be powered by its associated diesel generator from the electrical distribution system.

During normal plant operation, the containment atmosphere is inerted with dry nitrogen, meaning no external air is introduced. Nitrogen gas is drawn from the upper containment and CRD area, cooled by the AHU coils, and then distributed through supply ducts to maintain target temperatures.

During outages requiring containment access, outside air is introduced via the HVAC through the CIS interface. The CCS AHUs then recirculate this mixed air to:

- control oxygen levels
- support heat removal
- maintain safe temperatures for personnel entry

The CCS AHUs and chillers are backed up by diesel generators, ensuring continued operation during a LOPP. The CCS helps prevent the containment environment from exceeding design limits and supports cooldown from hot shutdown to cold shutdown following a LOPP.

PSAR Section 9.4.7 states that the containment cooling system conforms to the requirements of GDC 2. The NRC staff based its review of containment cooling system compliance with GDC 2 requirements consistent with RG 1.29, Regulatory Position C.2. Based on its review of the PSAR, the NRC staff understands that the CCS is not credited for mitigation of any DBAs and the CCS cooling functions are SC3/SCN and the principal components of the system are categorized as Seismic Category II. The NRC staff notes that the containment internal structures is categorized as Seismic Category I.

RG 1.29 C.2 indicates “Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level, or could result in incapacitating injury to occupants of the CR, should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.”

The applicant stated in PSAR Section 3.1, that SSCs whose failure during a seismic event could prevent a Seismic Category I SSC from performing its Safety Category 1 functions are designed to ensure such failure does not occur. Therefore, the NRC staff finds that the guidance of RG 1.29 has been satisfied and therefore concludes that the proposed design is consistent with the requirements of GDC 2 as it relates to CCS. A more detailed evaluation of seismic analysis of this system is contained in Chapter 3 of this report.

9.4.7.4 Conclusion

The NRC staff review concludes that the applicant's preliminary containment cooling system design for CRN-1 is consistent with the NRC regulations as set forth in Appendix A, GDC 2, for seismic qualification. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

9.5.1.1 Introduction

The FPS is part of the Fire Protection Program (FPP) which employs the defense-in-depth philosophy to provide assurance that the plant can achieve and maintain a safe shutdown condition and minimize radioactive releases to the environment in the event of a fire. The above safety objectives are achieved by installing and implementing an FPS/FPP to (1) prevent fires from starting, (2) detect, rapidly control, and promptly extinguish those fires that do occur, and (3) provide protection for SSC important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

The primary components of the FPS include:

- fire and smoke detection systems
- automatic and manual fire suppression systems
- rated fire barriers including walls, ceilings, floors, and opening seals such as fire doors, fire/smoke dampers, penetration seals, etc.

9.5.1.2 Regulatory Evaluation

The applicable regulatory requirement for the evaluation of fire protection at the CP stage is 10 CFR 50.34(a)(3) (TN249). 10 CFR Part 50, Appendix A, Criterion 3, "Fire Protection," is the design criteria relevant to fire protection and the applicant has stated in PSAR Section 3.1.1.3 that it meets this Criterion.

The NRC staff also finds the following regulations applicable to the design:

10 CFR 50.34, "Contents of applications; technical information," including:

10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."

10 CFR 50.35.

10 CFR Part 50, Appendix A, Criterion 3, “Fire protection,” states:

> Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

The NRC staff reviewed the information in the PSAR regarding the FPS to ensure that it provides reasonable assurance that the final design will conform to the design bases for fire protection. The applicable regulatory requirements relevant to the FPS at the OL stage include 10 CFR 50.48(a). The NRC staff did not make determinations regarding the OL requirements. The NRC staff’s review ensures that there is reasonable assurance that the preliminary CP information would support compliance with the applicable requirements at the OL stage. The applicant stated that the FPS of the CRN-1 Site is designed to comply with NRC Regulatory Guide (RG) 1.189, “Fire Protection for Nuclear Power Plants,” Revision 4, May 2021, and National Fire Protection Association (NFPA) 804, “Standard for Advanced Light Water Reactor Electric Generating Plants, 2020.” The NRC staff considered this guidance in its review, to determine whether there was reasonable assurance that the preliminary CP information would support compliance with the applicable requirements at the OL stage.

9.5.1.3 Technical Evaluation

The NRC staff reviewed the information in PSAR Section 9.5.1 regarding the FPP and the FPS. The NRC staff also reviewed PSAR Appendix 9A, “Fire Hazards Analysis (FHA).” The applicant stated in PSAR Section 9.5.1.2.1, “Fire Protection Program General Information,” that the FPS design follows the applicable guidance of RG 1.189, Revision 4, and NFPA codes and standards as applicable.

The applicant stated in PSAR Section 9.5.1.1 that the design bases of the FPS include ensuring one redundant division necessary to achieve fire safe shutdown is free of fire damage. RG 1.189, Regulatory Position 5.3, provides one means to ensure one safe shutdown success path is free of fire damage for a single fire in any single fire area which is to provide 3-hour rated fire barriers between redundant safe shutdown systems, components, and cables. However, PSAR Section 9.5.1.2.3, “Plant Arrangement,” stated that “3-hour rated barriers including walls, floors, and ceilings have fire resistance ratings of three hours where required based on high combustible loadings in the room (e.g., lubrication oil tank),” and for the electrical rooms and battery rooms, “fire barrier rating, fire suppression type, and fire detection for these rooms are defined by the Fire Hazards Analysis.” The NRC staff requested clarification as to whether 3-hour rated fire barriers are limited to rooms with high combustible loadings or are used throughout to separate redundant safe shutdown paths regardless of the combustible loading in the rooms.

In a supplement submittal letter dated December 2, 2025 (TVA 2025-TN13047), Enclosure 2 (TVA 2025-TN13045), the applicant stated, "the walls, floors, and ceilings have fire resistance ratings of three hours where required when an adjacent room contains equipment or systems necessary to ensure at least one fire safe shutdown path is available. This also includes fire doors, fire dampers, and penetration seals to provide separation of fire safe shutdown paths. Alternative fire safe shutdown features are provided where separation by three-hour barriers is not possible (e.g., main CR and secondary control room). Equivalency evaluations may be required during detailed design (for example, of a 3-hour rated fire area boundary). These evaluations are performed by a fire protection engineer (assisted by others as needed) with justifications retained as part of the fire protection report at the OL stage." The NRC staff concluded that the supplemental information clarified that design will conform to the guidance of RG 1.189. Based on the review described above, the NRC staff concludes that the additional information provided by the applicant regarding fire barriers is sufficient and and that the proposal is consistent with GDC 3.

The NRC staff reviewed the description of the detection and suppression systems of the FPS at CRN Site and determined that proper NFPA standards were invoked for the installation, inspection, testing and maintenance of the FPS consistent with RG 1.189. Therefore, it is consistent with GDC 3.

The applicant stated in PSAR Section 9.5.1.1 that the design bases of the FPS include ensuring one redundant division necessary to achieve fire safe shutdown is free of fire damage. Regulatory Guide 1.189, Regulatory Position 6.1.8, diesel generator rooms, provides one means to limit damages to safe shutdown equipment due to fire propagation by ensuring that diesel generator day tanks should be in a separate enclosure with a fire resistance rating of at least 3 hours, including doors or penetrations. These enclosures should be capable of containing the entire contents of the day tanks and should be protected by an automatic fire suppression system; or in the diesel generator room in a diked enclosure that has sufficient capacity to hold 110 percent of the contents of the day tank or is drained to a safe location.

In PSAR Section 9.5.4.2, TVA stated that the SDG are self-contained skid mounted power packages and are located in the diesel generator rooms. However, there was no discussion on how a postulated oil spill is contained. The NRC staff requested clarification on the design features of the generator rooms for the mitigation of postulated oil spills.

In a supplement submittal letter dated December 2, 2025 (TVA 2025-TN13047), Enclosure 2 (TVA 2025-TN13045), the applicant further provided that the SDG rooms contain automatic fire suppression, and the SDGs are separated from each other and from other areas of the plant by fire barriers that have a fire resistance rating of at least 3 hours. In addition, the SDG day tanks are double-walled to provide containment and minimize the chance of a fuel spill in the SDG rooms. The applicant also commits to performing a Fire Hazard Analysis and Fire Safe Shutdown Analysis in accordance with RG 1.189, Revision 4, to demonstrate that the plant can maintain the ability to perform fire safe shutdown functions and limit radioactive material releases to the environment in the event of a fire. Based on the review described above, the NRC staff concludes that the additional information provided by the applicant regarding the SDGs is sufficient and consistent with applicable guidance and, therefore, with GDC 3.

In PSAR Section 9.5.1.9, Exceptions and Alternative Requirements, the applicant took one exception to RG 1.189, Position 3.2.1(j), which states "Provisions should be made to supply water to at least two standpipes and hose connections for manual firefighting in areas containing equipment required for safe plant shutdown in the event of a SSE. The piping system serving such hose stations should be analyzed for SSE loading and should be provided with supports to ensure system pressure integrity.” Contrary to the guidance in Position 3.2.1(j), the piping system serving the standpipe system is designated as Seismic Category Non-Seismic, thus water supply to the standpipe system is not assured following an SSE event. As justification, the applicant stated that the CRN-1 proposed design will achieve the fire protection goals by providing the ability to safely shut down the plant in the event of an SSE and a subsequent fire in the RB through the use of fire barriers having a fire rating of 3 hours between fire safe shutdown SSC paths, and, therefore, the use of sprinklers or manual firefighting is not necessary to achieve safe shutdown. The applicant further stated that this will be demonstrated in the FSSA where the worst-case postulated fire is assumed to completely burn out the designated compartment without the application of fire suppression water. The applicant also stated that the FHA and FSSA will be completed for CRN-1 prior to issuance of the FSAR.

The applicant’s proposed exception could be acceptable provided the FHA and FSSA demonstrate that the FPS design basis function (ensuring one redundant division necessary to achieve fire safe shutdown is free of fire damage) is met. The NRC staff will review the completed FHA and FSSA in the OLA to ensure that the requirements of GDC 3 and 10 CFR 50.48 are met in the event of a SSE and subsequent fire. The NRC staff concludes that the information provided by the applicant regarding the exception to RG 1.189, Position 3.2.1(j), is sufficient and could be consistent with the applicable guidance and, therefore, for the purposes of CP issuance, is consistent with GDC 3.

9.5.1.4 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR Section 9.5.1 is sufficient and is consistent with the applicable guidance and regulatory requirements described in this section. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.5.2 **Non-Distributed Control and Information Communication System**

9.5.2.1 Introduction

The NRC staff understands that the information in the PSAR Section 9.5.2, Non-Distributed Control and Information Systems (N-DCIS), is consistent with the information the NRC staff would evaluate using to Section 9.5.2 of the SRP, “Communications Systems”, which is the guidance the NRC staff used to review this section of the PSAR.

Additional relevant information provided by the applicant for its communications systems is located in its PSAR in Section 1.9, “Regulatory Conformance,” Section 3.1, “Compliance with U.S. Nuclear Regulatory Commission General Design Criteria” and Section 13.3, “Emergency Preparedness.”

The review focuses on comparing the defined attributes of both the intra-plant and plant-to-offsite communications systems used for normal, abnormal, and emergency operating conditions. These communications systems are for both alerting plant personnel (intra-plant) and emergency response organizations, including law enforcement (plant-to-offsite) to ensure proper notification of all affected stakeholders in relevant situations.

The PSAR, Section 9.5.2, describes the N-DCIS as performing SC-3 functions (except the Radio System, which is classified as SC-N, as described in PSAR Table 3A-1, "Preliminary BWRX-300 Component Classification List." These communications support the coordination between plant personnel during normal, transient, abnormal operational occurrences and events or incident conditions, including transients, fire, security events, and accident conditions. This includes providing effective communication to appropriate plant staff under maximum noise levels and offsite personnel in the presence of a loss of preferred power events.[5]

9.5.2.2 Regulatory Evaluation

The regulatory evaluation is based on the applicant meeting, or planning to meet or satisfy relevant regulations. The NRC staff has also listed the guidance documents.

Regulations

- 10 CFR 50.34(a) Preliminary Safety Analysis Report (TN249)
  - 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.
  - 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.
  - 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
  - 10 CFR 50.34(a)(10) requires discussion of the applicant's preliminary plans for coping with emergencies.
- 10 CFR 50.34(f)(2)(xxv), "Emergency Response Facilities," (TMI Action Plan Item III A.1.2).
- 10 CFR 50.35.
- 10 CFR 50.47(b)(8)
- 10 CFR 50.55a, "Codes and Standards."
- 10 CFR Part 50, Appendix A, "Generic Design Criteria for Nuclear Power Plants"
  - General Design Criterion (GDC) 1, "Quality Standards and Records."
  - GDC 2, "Design Basis for Protection Against Natural Phenomena."
  - GDC 3, "Fire Protection."
  - GDC 4, "Environmental and Dynamic Effects Design Bases."
  - PDC 19, "Control Room."
- 10 CFR Part 50, Appendix E "Emergency Planning and Preparedness for Production and Utilization Facilities," (particularly part IV.E(9), as it relates to the provision of at least one onsite and one offsite communications system, each with a backup power source).

[5] Some of the regulatory requirements cited in Section 9.5.2 of the SRP are not directly applicable to an application for a CP, and the NRC staff's analysis in this section does not directly cite to or assess the applicant's satisfaction of those requirements.

Guidance

- RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," Revision 3, November 1978 (NRC 1978-TN12879)
  - Section 3.1, "Conformance with NRC General Design Criteria"
  - Section 9.5.2, "Communications Systems"
- Chapter 9, Auxiliary Systems, Section 9.5.2, "Communications Systems", of NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Revision 3, March 2007 (NRC 2021-TN8013)
- DNRL-ISG-2022-01 – Safety Review of Light-Water Power-Reactor Construction Permit Applications (NRC 2022-TN12894)

9.5.2.3 Technical Evaluation

10 CFR 50.34(a)(3)(i) and 50.34(a)(3)(ii) require an applicant's PSAR to contain the minimum listed technical information, including the preliminary design of the facility, which must include the principal design criteria (PDC) and design bases and explain how the PDC and the design bases interrelate with each other.

Section 9.5.2 describes the NRC staff's evaluation of the technical information provided by the applicant in relation to the important communications systems located throughout the power plant and those required external communications systems and facilities during normal, abnormal, and emergency conditions. This technical information is provided in accordance with the applicable portions of 10 CFR 50.34 that are referenced in this section. Additionally, as these systems must be deigned to be fully operational, (including reliable back-up power requirements), during emergency conditions, whether due to natural phenomena, (which would require compliance with general design criteria (GDC) 2, 3, and 4), or all types of evaluated plant events, (i.e., design basis and BDB events as directed by Commission Policy), 10 CFR 50.47(b)(8), 10 CFR 50.55a, and GDC1 would also apply. Also, given that the nexus for these normal, abnormal, and emergency communications typically emanates in, from, or to the main control room (MCR), PDC 19 would apply.

Examining the relevant information in Table 1.9-9, associated with the acceptance criteria related to PSAR Section 9.5.2, the applicant chose to delay conformance with 10 CFR 50.34(f)(2)(xxv)

10 CFR 50.34(f)(2)(xxv), states, "Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near site Emergency Operations Facility. (III.A.1.2)[6]", the table notes that the information related to the conformance of and for these facilities will be presented in the FSAR.

As it relates to the emergency communications and related facilities that support them, a further review of the references in Section 9.5.2 of the PSAR, specifically Section 13.3, "Emergency Preparedness", of Chapter 13, "Conduct of Operations" is associated with the Technical Support Center (TSC), the Operational Support Center (OSC), and the Emergency Operations Facility (EOF). Specifically, Section 13.3 of the PSAR states, "Section 13.3 of the Early Site Permit Application (ESPA) Site Safety Analysis Report (SSAR), in conjunction with Part 5 Emergency

[6] The information in the parenthetical refers to the specific section of the related Three Mile Island (TMI) Action Plan Task.

Plan of the ESPA, describes emergency preparedness for a small modular reactor (SMR) facility at the Clinch River Nuclear (CRN) Site."

In addition, Section 13.3 of the PSAR states,

- Part 5A of the ESPA contains the major features of an emergency plan for a PEP EPZ at the site boundary of the CRN Site.
- Part 5B of the ESPA contains the major features of an emergency plan for a PEP EPZ that consists of an area approximately 2 mi in radius from the center point of the CRN Site.

  Each of the major features of the emergency plan complies with 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50, except where exemptions were requested, as described in Part 6, "Exemptions and Departures," of TVA's ESPA.

From a review of the material in the Part B, "Emergency Plan" (2-Mile EPZ), Revision 1" (TVA 2017-TN5442), the following information was located:

- **Section 2.4**, "National Response Framework", states that TVA will utilize the Central Emergency Control Center (CECC) in Chattanooga Tennessee as the Emergency Operations Facility (EOF) as defined for the CRN Site.
- **Section 6.0**, "Communications", describes how the plant telephone switching equipment consists of one or more switching centers equipped with fully redundant common logic and redundant power sources, which would satisfy the requirement for the onsite to possess a back-up power source, provided there is a sufficient level of independence between the two power sources, such that the failure of one power system would not impact the alternate, back-up power supply.
  - The Offsite Communications Equipment subsection describes that multiple systems perform their different communications function(s) with redundant and reverse routes, and that the systems will operate in a reliable manner, however the level of detail describing "how" the various systems' redundant and backup power supplies operate is beyond the scope of information that would be expected at the PSAR stage of licensing.
- **Section 8.0** describes the functions and attributes of the TSC, OSC, and the CECC, (that will serve as the EOF for the CRN Site).

From a review of the information in Section 9.5.2 and Section 13.3 of the PSAR, and Part B, "Emergency Plan" (2-Mile EPZ), Revision 1, referenced in Section 13.3 of the PSAR, the NRC staff determined that, while there is no information in these documents that explicitly states that all necessary external communications systems will possess a backup power source and other specific reliable functionality, it is acceptable for the applicant to provide that level of detailed information in the FSAR (or demonstrate in the FSAR that there are multiple, redundant communications systems that are capable of performing the necessary offsite communications systems' functions).10 CFR Part 50, Appendix E, part IV.E(9) provides in pertinent part that an applicant's emergency plan must make and describe adequate provisions for emergency facilities and equipment that include at least one offsite communications system, and it requires that any offsite communications system have a backup power source. But Appendix E also makes clear that these are minimum requirements for the emergency plans that must be "submitted as part of the [FSAR] for an operating license" and that an applicant's emergency plans need only be "described generally" (and not submitted) in the PSAR for a CP. Accordingly, the NRC staff has concluded that these details, including information regarding backup power systems for any offsite communications systems to be included in the applicant's emergency plan, need not be included in the general description of emergency plans submitted by the applicant at this CP stage. The applicant will need to include these details in its emergency plan submitted as part of the FSAR for an OL, and the NRC staff reasonably expects this information will be available and sufficient to evaluate prior to the completion of construction.

10 CFR 50.34(f)(2) states that the information called for by 10 CFR 50.34(f)(2) is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. 10 CFR 50.35(a)(2) specifically refers to "such further technical or design information as may be required to complete the safety analysis, and which can reasonably be left for later consideration." Given that reference to 10 CFR 50.34(f)(2) and the fact that the emergency plan does provide some detailed information about the TSC, OSC, and the nearsite EOF, the NRC staff has determined that this information is sufficient for the CP stage and, therefore, consistent with 10 CFR 50.34(f)(2)(xxv).As noted elsewhere in this SE, the applicant is complying with a custom PDC 19 rather than with GDC 19. The NRC staff reviewed the use of PDC 19 in section 6.4 and chapter 7 of this report. The NRC staff concluded it was acceptable. With regard to the communications systems being reviewed here, based on the above information, , the NRC staff determined that the level of information describing both onsite and offsite communications systems in the CR that is necessary at the CP stage, is acceptable to the NRC staff based on the totality of the information from the PSAR and Part B of the Emergency Plan. Further, the NRC staff concluded that, based on the information in the emergency plans, the submission is consistent with GDCs 2, 3, and 4. This is because, with regard to plant telephone switching equipment consists of one or more switching centers equipped with fully redundant common logic and redundant power sources and with regard to the offsite communications equipment multiple systems perform their different communications function(s) with redundant and reverse routes, and that the systems will likely operate in a reliable manner. If implemented as described, this would allow the communication system to withstand the events described in GDCs 2, 3, and 4.

Further, as noted above, the applicant stated that the emergency plan complies with 10 CFR 50.47(b)(8). Given that the applicant need only provides preliminary plans for coping with emergencies at the CP stage and the above-discussed language from Appendix E to 10 CFR Part 50, the NRC staff concluded that this is sufficient for the CP stage.

With regard to 10 CFR 50.55a and GDC 1, the NRC staff concluded that the submittal is consistent with 10 CFR 50.55a and GDC because, as discussed above there are back-up systems which, as discussed elsewhere in the SE, may use codes referenced in 10 CFR 50.55a.

9.5.2.4 Conclusion

The NRC staff evaluated the CRN Site design, design criteria, and design bases to determine whether they conform to the regulations, staff positions and industry standards for communication systems. The information provided by PSAR and the Emergency Plan sufficiently described the Clinch River communication systems in accordance with the applicable regulations and guidance identified.

Based on its findings above, the NRC staff concludes the design of the Clinch River communication systems, as described in PSAR Sections 3.1, 9.5.2, and 13.3, along with the Clinch River Emergency Plan (2 mi EPZ), Revision 1, provides sufficient preliminary information and is consistent with the applicable regulatory requirements and guidance identified in this section. Therefore, the NRC staff concludes that the design of the communications systems is consistent with the NRC staff's review criteria and industry standards and finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

A more comprehensive staff evaluation of more detailed design information that the applicant will provide in its FSAR will occur during the review of the CRN OLA, at which time the NRC staff will verify that the final CRN design communication systems conform to its design basis.

9.5.3 Lighting

9.5.3.1 Introduction

SRP Section 8.5.3 states that lighting should provide adequate lighting during all plant operating conditions, including fire, transient and accident conditions. The lighting is comprised of normal lighting, emergency lighting, and CR lighting. PSAR Section 9.5.3.3 states that

> Lighting is not essential for reactor shutdown, containment isolation, or containment and reactor heat removal. Lighting is not essential in preventing significant release of radioactive material to the environment. Failure of normal and emergency lighting does not compromise automatic actuation of safety class components or systems that perform a Safety Category function. Emergency lighting and CR lighting provide illumination for operations during emergencies and anticipated operational occurrences.

9.5.3.2 Regulatory Evaluation

No specific GDC or other requirements directly apply to the performance of the lighting systems.

10 CFR 50.34(a)(2) (TN249) states that the PSAR should include a summary description, with attention to design and operating characteristics, unusual or novel design features, and principal safety considerations. The NRC staff also finds the following regulations applicable to the design:

10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."

10 CFR 50.35, "Issuance of construction permits." A CR design includes lighting for operators to perform actions, and NUREG-0700 provides detailed acceptance criteria for human factors engineering design attributes, including lighting.

The following NRC guidance applies to the review of lighting:

- SRP Section 9.5.3, Revision 3, "Lighting Systems," issued March 2007, provides acceptance criteria for the lighting systems. The NRC staff review of the lighting systems should determine that they (1) provide adequate lighting in all areas of the plant during normal plant operations, (2) provide adequate emergency lighting during all plant operating conditions, including fire, transient, and accident conditions, (3) address the effect of the loss of all alternate current (AC) power (i.e., during an SBO) on the emergency lighting system. Further, one of the acceptance criteria is that the control room lighting systems have adequate illumination levels that conform to the illumination levels recommended in NUREG-0700.

- NUREG-0700, Revision 3, “Human-system Interface Design Review Guidelines,” as it relates to acceptable lighting levels (NRC 2019-TN6347).
- RG 1.75, Revision 3, applies as it relates to the physical separation and electrical isolation that must occur between safety-related and not safety-related circuits to maintain the independence of safety-related circuits and equipment so that the safety functions required during and following any DBE can be accomplished (NRC 2005-TN13229).
- RG 1.189, Revision 4, Regulatory Position 4.1.6, “Emergency Lighting”, as it relates to emergency lighting for post-fire safe shutdown (NRC 2021-TN9419).

9.5.3.3 Technical Evaluation

The NRC staff reviewed the information in PSAR Section 9.5.3, to determine whether the plant lighting systems provide adequate lighting during all plant operating conditions and whether the lighting systems can operate without adversely impacting the operation, control, and maintenance of safety-related SSCs.

Normal Lighting

SRP Section 9.5.3 states that one of the areas of review is the capability the normal lighting systems to provide adequate lighting during all plant operating conditions. PSAR Table 3A-1 states that the normal lighting is non-safety (i.e., SCN). PSAR Section 9.5.3.2 states that the preferred power system provides power to normal lighting. PSAR Section 9.5.3.3 states that normal lighting provides illumination throughout the plant, and normal lighting is available under normal plant operating, maintenance, and testing conditions. PSAR Section 9.5.3.3 states that the plant lighting conforms to NUREG-0700 as it relates to lighting level recommendations. Since the normal lighting provides lighting during all plant conditions and conforms to NUREG-0700 illumination levels, the NRC staff finds the normal lighting is consistent with SRP Section 9.5.3 and is, therefore, acceptable.

Emergency Lighting

SRP Section 9.5.3 states that one of the areas of review is the capability of the emergency lighting system to provide

adequate lighting during all plant operating conditions, including fire, transient, and accident conditions. Specifically, the emergency lighting should provide adequate emergency station lighting in all areas, required for firefighting, control and maintenance of equipment used for implementing safe shutdown of the plant during all plant operating conditions, and the access routes to and from these areas. PSAR Table 3A-1 states that emergency lighting has a SC3 safety classification and location in SC1 structures. PSAR Section 9.5.3.2 states that

> Emergency lighting provides acceptable levels of illumination when normal lighting is not available. Emergency lighting is available in areas where emergency operations are performed, including the access and egress routes to and from those areas, in response to a fire, and safe shutdown. All emergency lighting derives power from a source backed by standby power. Lighting supporting egress and safe shutdown is maintained by standby power during LOPP and includes or is supplemented by self-contained battery pack units that provide power for sufficient lighting during a station blackout. The

self-contained battery pack units are inspected and tested in accordance with manufacturer recommendations. For safe shutdown, access, and egress to safe shutdown areas, self-contained 8-hour battery pack units are provided for general egress, and life safety purposes to meet building codes. Self-contained 90-minute battery pack units are provided.

PSAR Section 9.5.3.3 further states that the plant lighting conforms to NUREG-0700, as it relates to lighting level recommendations. Since the emergency lighting is powered by standby power and self-contained battery packs and conforms to NUREG-0700 illumination levels, the NRC staff finds the emergency lighting is capable of providing adequate lighting during all plant condition, consistent with SRP Section 9.5.3, and is acceptable.

RG 1.189, Regulatory Position 4.1.6, “Emergency Lighting” states that “Emergency lighting should be provided throughout the plant as necessary to support fire suppression actions and safe-shutdown operations, including access and egress pathways to safe-shutdown areas during a fire event.” Specifically for post-fire safe shutdown, RG 1.189, Regulatory Position 4.1.6.2a states that “fixed, self-contained lighting consisting of units with individual 8-hour minimum battery power supplies should be provided in areas needed for operation of safe-shutdown equipment and for access and egress routes to these areas.” Since the emergency lighting contains self-contained battery pack units for access, egress, and safe shutdown activities, the NRC staff finds that the emergency lighting is consistent with RG 1.189, Regulatory Position 4.1.6. Although RG 1.189 provides an acceptable approach to addressing 10 CFR 50.48(a) and (b), it contains information on emergency lighting which is relevant for the cited regulatory requirements. As this approach is acceptable for meeting more specific regulatory requirements, consistency with this approach is one way to meet the relevant requirements in 50.34(a).

SER Section 8 provides the NRC staff’s evaluation of electric power, including the standby power system. The NRC staff finds the emergency lighting is consistent with the guidance in SRP Section 9.5.3 since the emergency lighting system provides adequate station lighting in all vital areas from onsite standby power sources during the full spectrum of accident and/or transient conditions and to the access routes to and from these areas. In addition, the NRC staff finds that the applicant has addressed the effect of the loss of all ac power (i.e., during an SBO) on the emergency lighting system because standby power supports emergency lighting and self-contained battery pack units are available. Section 8.4 of this SER provides the NRC staff’s review and evaluation of an SBO.

Control Room Lighting

SRP Section 9.5.3 states that the adequacy of CR lighting systems and features related to their effectiveness to support reliable human performances should be evaluated with respect to the criteria specified in NUREG-0700. PSAR Table 3A-1 states that CR lighting has a SC2 safety classification and is located in the RB, a SC1 structure, and CB, a SC2 structure. PSAR Section 9.5.3.2 states that lighting for the MCR and SCR provides acceptable illumination under normal and emergency conditions and is powered from two divisions of the emergency power system which are backed by 72-hour seismically qualified batteries. Section 8.3.2 of the SER provides the NRC staff’s review of the emergency power system and finds that it is consistent with RG 1.75 as it relates to the physical separation and electrical isolation that must occur between safety-related and non-safety-related circuits to maintain the independence of safety-related circuits and equipment so that the safety functions required during and following any DBE can be accomplished. PSAR Section 9.5.3.3 states that CR lighting features comply with

NUREG-0700, as it relates to CR lighting level recommendations. The NRC staff will review CR lighting levels to verify consistency with NUREG-0700 at the OL stage. Not reviewing this now is acceptable because this cannot be verified until lighting has been installed. Since the MCR and SCR lighting is powered by the emergency power system for a minimum of 72 hours and conforms to NUREG-0700 illumination levels, the NRC staff finds the MCR and SCR lighting is consistent with SRP 9.5.3 and is, therefore, acceptable.

Inspection and Testing of Lighting Systems

PSAR Section 9.5.3.2 states that the self-contained battery pack units in the emergency lighting system are inspected and tested in accordance with manufacturer recommendations. PSAR 9.5.3.4, "Inspection and Testing," states that preoperational testing of lighting verifies that the normal lighting system provides illumination under normal plant operating, maintenance, and testing conditions, and that the emergency lighting system provides illumination throughout the station, including areas where emergency operations are performed. Inspection and testing of normal and emergency lighting are addressed as part of preoperational testing. Section 14.2 of the SER provides the associated staff evaluation of preoperational testing. Since the self-contained battery pack units are inspected and tested in accordance with manufacturer recommendations and preoperational testing verifies the illumination of the normal, emergency, and CR lighting, the NRC staff finds the inspection and testing conforms to SRP Section 9.5.3.

9.5.3.4 Conclusion

The NRC staff reviewed the normal and emergency lighting and CR lighting for conformance with the guidelines of SRP Section 9.5.3, NUREG-0700, and RG 1.189, Revision 4. Based on the above technical evaluation, the NRC staff concludes that the normal lighting, the emergency lighting, MCR and SCR lighting designs provide adequate illumination in all areas of the plant and access routes to these areas under all plant operating conditions such as normal, transient, fire, accident, and SBO conditions, as recommended by SRP Section 9.5.3. The NRC staff finds that the information on lighting systems in the PSAR is consistent with the requirements in 10 CFR 50 34.(a)(2) and (3)(iii) and guidance identified in this section. Therefore, is sufficient and adequately supports the issuance of a CP pursuant to the regulations under 10 CFR 50.35

9.5.4 Standby Diesel Generator Fuel Oil Storage and Transfer System

9.5.4.1 Introduction

This section discusses the SDG fuel oil storage and transfer system which is a separate equipment and piping system used to supply diesel fuel to the SDGs. The SDGs provide electrical power to the Standby Power System as described in Chapter 8 of the PSAR.

9.5.4.2 Regulatory Evaluation

The NRC staff reviewed the proposed design in accordance with SRP Section 9.5.4, Revision 3, March 2007 (NRC 2021-TN8013). The following regulatory requirements apply:

- GDC 2, as it relates to the system's capability to withstand the effects of natural phenomena, including earthquakes and tornadoes.
- GDC 4 as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of externally- and internally- generated missiles, pipe whip, and jet impingement forces of pipe breaks.

- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions.
- GDC 17, as it relates to the capability of the diesel engine fuel oil system to meet independence and redundancy criteria.

The NRC staff also finds the following regulations applicable to the design:

- 10 CFR 50.34, "Contents of applications; technical information," including (TN249):

  – 10 CFR 50.34(a)(3)(i) requires that the PSAR include the principal design criteria (PDCs) for the preliminary design of the facility.

  – 10 CFR 50.34(a)(3)(ii) requires that the PSAR describe the design bases and the relation of the design bases to the PDCs.

  – 10 CFR 50.34(a)(3)(iii) requires that the PSAR include "Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety."
- 10 CFR 50.35.

9.5.4.3 Technical Evaluation

The diesel fuel oil storage and transfer system provide storage and transfer of diesel fuel oil to the day tanks of the SDGs. This subsystem is comprised of one (1) 30,000 gallon above-ground diesel fuel oil storage tank, two (2) fully redundant diesel fuel oil transfer pumps, and associated piping, valves, and instrumentation. The pumps are located outside of the protected area and transfer fuel oil underground from the diesel fuel oil storage tank to the SDG day tanks located in the diesel generator rooms. Figure 9.5-2 "Diesel Fuel Oil Storage and Transfer System Simplified Diagram" provides a diagram of the system and main components.

The storage tank is situated in the yard and holds enough fuel for a minimum of 7 days of SDG operation in case of a loss of offsite power. The SDG fuel handling system is used to support the SDG operation discussed in PSAR Section 8.3.1 that describes the required time of SDG operation as, "Once started, operate without requiring any plant services other than those required to store and transfer fuel, with the capability of continuous operation for at least 7 days at rated power without any offsite resources."

The NRC staff evaluated the application's consistency with the GDCs listed above. The applicant indicated in PSAR Section 9.5.4 and PSAR Table 1.9-9 that compliance with the GDCs is not applicable based on the design not including SC1 diesels.

The SRP 9.5.4 provides GDC 17 as applicable design criteria. However, the applicant has proposed PDC 17, as discussed in section 8.1.3 of this report. PSAR Section 8.1.3 states the standby diesel generators are not required for 72-hours post DBA. The SDGs are not credited in safety analysis and do not have safety-related functions. Therefore, the SDG fuel oil system does not have any safety-related functions. The NRC staff agrees that PDC 17 is not applicable to the SDG fuel oil system.

GDC 2 and 4 are related to the requirements to withstand natural phenomena and pipe failure. However, PSAR Section 9.5.4.3 indicates that GDC 2 and 4 are not applicable. The PSAR indicates that SDG fuel oil system is designed as a SC3 support system and failure of the system does not impact SC1 SSCs. The PSAR further indicates the SSC for the system are also located outside the protected area which is away from safety-related SSCs. PSAR Section 9.5.4.2 indicates tanks are provided with berms capable of holding and monitoring contents from leakage and underground fuel oil lines are designed to allow for leak detection. Therefore, the NRC staff has confidence that any oil leakage or pipe break can be detected and is located away from safety related equipment. In addition, the SDGs are located in the turbine building and PSAR Section 3.5.1.1 indicates the diesel generator equipment is examined for possible sources of creditable and significant missiles. Based on this being support system for SC3 SDG, the NRC staff agrees with applicant that GDC 2 and 4 are not applicable for the described fuel oil system. Additional review of flood protection and system location may be required at OL stage to verify other systems piping failure will not result in adverse condition to the SDG function.

The Clinch River Nuclear (CRN) Site is a single unit, single-module site, therefore no SSCs important to safety are shared between units or modules. Therefore, the SDG design can be in conformance with GDC 5. The NRC staff finds that the level of detail provided on the SDG fuel oil support system sufficient to demonstrate an adequate design basis for a preliminary design during the CP review. A more detailed evaluation of information will occur during the review of the Clinch River OLA, at which time the NRC staff will confirm GDC applicability to the SDG support system.

9.5.4.4 Conclusion

The NRC staff evaluated the design features of the SDG fuel oil storage and transfer system in PSAR Section 9.5.4. Based on the NRC staff's review and the findings discussion above, the NRC staff concludes that the preliminary design of the SDG fuel oil storage and transfer system is consistent with the applicable regulatory requirements described in this section. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.

9.5.5 Standby Diesel Generator Cooling Water System

The SDG is a packaged system and details will be provided in the FSAR. The SDGs are radiator cooled and do not require additional mechanical support for cooling such as an auxiliary cooling water system. This system supports the SDG operation discussed in PSAR Section 8.3.1.

The NRC staff review of the SDG is limited because the PSAR does not contain a detailed description of the SDG support system. Since the SDG is not credited to support core cooling or containment integrity, an evaluation of the system design is not required for the issuance of a CP.

A more detailed evaluation of the SDG will occur during the review of the Clinch River OLA, at which time the NRC staff will confirm that the final design conforms to applicable GDCs.

9.5.6 Standby Diesel Generator Starting System

The applicant described the SDG as a packaged system and indicated details would be provided in the FSAR. PSAR Section 9.5.6 indicates the SDGs are electric start and do not require additional mechanical support for starting such as an air system. SDG starting time is not material to it performing its safety category functions. The SDGs may be started from the MCR, SCR, or a local panel. This system supports the SDG operation discussed in PSAR Section 8.3.1.

The NRC staff review of the SDG is limited because the PSAR does not contain a detailed description of the SDG support system. Since the SDG is not credited to support core cooling or containment integrity, an evaluation of the system design is not required for the issuance of a CP.

A more detailed evaluation of this system will occur during the review of the Clinch River OLA, at which time the NRC staff will confirm that the final design conforms to applicable GDCs.

9.5.7 Standby Diesel Generator Lubrication System

The applicant described SDG as a packaged system, and details will be provided in the FSAR. PSAR Section 9.5.7 indicates that once started, the SDGs operate without requiring plant services other than those required to store and transfer fuel, with the capability of continuous operation for at least 1 week. This system supports the SDG operation discussed in PSAR Section 8.3.1. The NRC staff review of the SDG is limited because the PSAR does not contain a detailed description of the SDG support system. Since the SDG is not credited to support core cooling or containment integrity, an evaluation of the system design is not required for the issuance of a CP.

A more detailed evaluation of this system will occur during the review of the Clinch River OLA, at which time the NRC staff will confirm that the final design conforms to applicable GDCs.

9.5.8 Standby Diesel Generator Combustion Air Intake and Exhaust System

The applicant described SDG as a packaged system, and details will be provided in the FSAR. PSAR Section 9.5.8 indicates that air is supplied from intakes located outside the building containing the SDGs. The SDG rooms have separate exhausts for radiator heat removal and combustion air removal. The exhaust for combustion air is routed to a safe discharge location. This supporting system is used to support the SDG operation discussed in PSAR Section 8.3.1. The NRC staff review of the SDG is limited because the PSAR does not contain a detailed description of the SDG support system. Since the SDG is not credited to support core cooling or containment integrity, an evaluation of the system design is not required for the issuance of a CP.

A more detailed evaluation of this system will occur during the review of the Clinch River OLA, at which time the NRC staff will confirm that the final design conforms to applicable GDCs.

9.5.9 Cranes, Hoists, and Elevators

PSAR Section 9.5.9 indicates that the facility is provided with several cranes, hoists, and elevators (CHE) throughout the plant which provide means to move equipment and materials along safe load paths. The PSAR does not include a description, location, or analysis of these CHE, but it indicates that additional details are to be provided in the FSAR. FSAR will include an evaluation of the CHE against the requirements of OHLHS program requirements discussed in PSAR Section 9.1.5, consistent with RG 1.244 (NRC 2021-TN13113). The NRC staff evaluated the OHLHS discussed in PSAR Section 9.1.5 and provided its evaluation in Section 9.1.5 of this report.

The PSAR indicates that the CHE does not support or provide Safety Category function, therefore they have no safety design basis.

The NRC staff reviewed the information provided in the PSAR and agreed that, since the CHE have no safety function, it is not required to be evaluated for the issuance of a CP.

Appendix 9A-Fire Hazards Analysis

9A.1 Introduction

PSAR Appendix 9A, “Fire Hazards Analysis”, describes the content and methodologies of the FHA and FSSA for the CRN Site. An FHA demonstrates that the plant can maintain the ability to perform fire safe shutdown functions and limit radioactive material releases to the environment in the event of a fire. While the CPA does not include all the analyses that are necessary for a complete FHA and FSSA, the applicant stated that the completed documents will be issued prior to the issuance of the FSAR.

9A.2 Regulatory Evaluation

The applicable regulatory requirement for the evaluation of fire protection at the CP stage is 10 CFR 50.34(a)(3) (TN249). 10 CFR Part 50, Appendix A, GDC 3, “Fire Protection,” is the design criteria relevant to fire protection and the applicant has stated in PSAR Section 3.1.1.3 that it meets this criterion. The NRC staff reviewed the information in the PSAR to ensure that it provides reasonable assurance that the final design will meet the relevant regulatory requirements for fire protection. The applicant stated that the FHA and FSSA will be developed utilizing the methodologies described in RG 1.189, “Fire Protection for Nuclear Power Plants,” Revision 4, May 2021, and National Fire Protection Association (NFPA) 804, “Standard for Advanced Light Water Reactor Electric Generating Plants, 2020.”.

9A.3 Technical Evaluation

The NRC staff reviewed the available information in the FHA with respect to the approach and methodology that will be applied in the final FHA. The applicant stated that the plant design utilizes the methodologies described in RG 1.189 and NFPA 804 for the development of the FHA and FSSA, which the NRC staff determined is sufficient and demonstrates consistency with GDC 3.

9A.4 Conclusion

Based on the review described above, the NRC staff concludes that the information in PSAR Appendix 9A could meet the requirements of GDC 3. Therefore, the NRC staff finds that the information provided by the applicant is sufficient and meets 10 CFR 50.34(a)(3) and adequately supports the issuance of a CP pursuant to the regulations in 10 CFR 50.35.