UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 24, 2021

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATION 3.7.2, "PLANT SERVICE WATER (PSW) SYSTEM AND ULTIMATE HEAT SINK (UHS)" (EPID L-2021-LLA-0164) (**EMERGENCY CIRCUMSTANCES**)

Dear Ms. Gayheart:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 311 to Renewed Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant (Hatch), Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 21, 2021, as supplemented by letter dated September 23, 2021.

The amendment revises TS 3.7.2, "Plant Service Water (PSW) System and Ultimate Heat Sink (UHS)," Condition A, "One PSW pump inoperable," to allow a one-time increase in the Completion Time (CT) from 30 days to 45 days.

The license amendment is issued under emergency circumstances as described in the provisions of paragraph 50.91(a)(5) of Title 10 of the *Code of Federal Regulations* due to the time critical nature of the amendment. The required CT for Condition A for TS 3.7.2 of 30 days is currently applicable and will expire on September 25, 2021, at 4:20 pm Eastern Time.

In this instance, an emergency situation exists, due to the failure of the 1C PSW pump post-maintenance test (PMT) on September 16, 2021, and Southern Nuclear Operating Co., Inc. (SNC) determined that both the pump and the motor would need to be replaced. SNC realized that the pump and motor could not be replaced in the 30-day CT and additional time would be needed. SNC did not foresee the motor failing during the PMT of the pump.

A copy of the related safety evaluation is also enclosed. The safety evaluation describes the emergency circumstances under which the amendment was issued and the final no significant hazards determination. A Notice of Issuance addressing the final no significant hazards determination and opportunity for a hearing associated with the emergency circumstances will be included in the Commission's biweekly *Federal Register* notice.

If you have questions, you can contact me at 301-415-3100 or at John.Lamb@nrc.gov.

Sincerely,

/RA/

John G. Lamb, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-321

Enclosures:
1. Amendment No. 311 to DPR-57
2. Safety Evaluation

cc: Listserv

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 311
Renewed License No. DPR-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 1 (the facility) Renewed Facility Operating License No. DPR-57 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated September 21, 2021, as supplemented September 23, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-57 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 311, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented immediately.

FOR THE NUCLEAR REGULATORY COMMISSION

Glenn E. Miller

Digitally signed by Glenn E. Miller
Date: 2021.09.24 17:34:26 -04'00'

Ed Miller, Acting Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-57
and Technical Specifications

Date of Issuance: September 24, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 311

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the license and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
4	4
TSs	TSs
3.7-3 3.7-4	3.7-3 3.7-4

for sample analysis or instrument calibration, or associated with radioactive apparatus or components

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

(C) This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 311, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance Requirement (SR) contained in the Technical Specifications and listed below, is not required to be performed immediately upon implementation of Amendment No. 195. The SR listed below shall be successfully demonstrated before the time and condition specified:

SR 3.8.1.18 shall be successfully demonstrated at its next regularly scheduled performance.

(3) Fire Protection

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 11, 2020. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

3.7 PLANT SYSTEMS

3.7.2 Plant Service Water (PSW) System and Ultimate Heat Sink (UHS)

LCO 3.7.2 Two PSW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PSW pump inoperable.	A.1 Restore PSW pump to OPERABLE status. OR ----------------NOTES--------------- 1. Only applicable during 1C PSW pump repair. 2. Only applicable until October 10, 2021 at 1620 EDT. --- A.2.1 Establish compensatory measures as described in letter NL-21-0862 dated September 23, 2021, Enclosure 5. AND A.2.2 Restore PSW pump to OPERABLE status.	30 days 30 days 45 days
B. One PSW turbine building isolation valve inoperable.	B.1 Restore PSW turbine building isolation valve to OPERABLE status.	30 days
C. One PSW pump in each subsystem inoperable.	C.1 Restore one PSW pump to OPERABLE status.	7 days
D. One PSW turbine building isolation valve in each subsystem inoperable.	D.1 Restore one PSW turbine building isolation valve to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 ------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	 12 hours
F. One PSW subsystem inoperable for reasons other than Conditions A and B.	----------------NOTES----------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for diesel generator made inoperable by PSW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by PSW System. --- F.1 Restore the PSW subsystem to OPERABLE status.	 72 hours
G. Required Action and associated Completion Time of Condition F not met. <u>OR</u> Both PSW subsystems inoperable for reasons other than Conditions C and D. <u>OR</u> UHS inoperable.	G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.	12 hours 36 hours

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 311 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-57

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-321

1.0 INTRODUCTION

By letter dated September 21, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21264A003), as supplemented by letter dated September 23, 2021 (ADAMS Accession No. ML21266A004), the Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted a license amendment request (LAR) for the Edwin I. Hatch Nuclear Plant (Hatch), Unit 1. The proposed amendment would revise the Hatch, Unit 1 Technical Specification (TS) requirements of TS 3.7.2, “Plant Service Water (PSW) System and Ultimate Heat Sink (UHS).” Specifically, the proposed amendment would revise Condition A, “One PSW pump inoperable,” to allow a one-time increase in the Completion Time (CT) from 30 days to 45 days.

The licensee requested U.S. Nuclear Regulatory Commission (NRC) approval of the proposed amendment in accordance with Section 50.91(a)(5) of Title 10 of the *Code of Federal Regulations* (10 CFR) regarding emergency situations, as discussed in Section 4.0 of this safety evaluation. The amendment would allow the licensee to have a 45-day CT for TS 3.7.2 for one time only.

2.0 REGULATORY EVALUATION

2.1 PSW System Design Overview

The licensee provided the following description of the PSW system in Section 2.2 of the Enclosure to the LAR:

> The PSW System is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs), residual heat removal (RHR) pump coolers, and room coolers for Emergency Core Cooling System equipment, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The PSW System also provides cooling to unit components, as required, during normal operation. Upon receipt of a loss of offsite power or loss of coolant accident (LOCA) signal, nonessential loads are

automatically isolated, the essential loads are automatically divided between PSW Divisions 1 and 2, and one PSW pump is automatically started in each division.

The PSW System consists of the ultimate heat sink (UHS) and two independent and redundant subsystems. Each of the two PSW subsystems is made up of a header, two 8500 gallons per minute (gpm) pumps, a suction source, valves, piping, and associated instrumentation.

Either of the two subsystems is capable of providing the required cooling capacity to support the required systems with one pump operating. The two subsystems are separated from each other so failure of one subsystem will not affect the operability of the other system.

Cooling water is pumped from the UHS (i.e., the Altamaha River) by the PSW pumps to essential components through the two main headers. After removing heat from the components, the water is discharged to the circulating water flume to replace evaporation losses from the circulating water system, or directly to the river via a bypass valve.

The ability of the PSW System to support long term cooling of the reactor containment is assumed in evaluations of the equipment required for safe reactor shutdown presented in the [Final Safety Analysis Report (FSAR), Section 10.7 (Reference 2). These analyses include the evaluation of the long-term primary containment response after a design basis LOCA.

The ability to provide onsite emergency AC [alternating current] power is dependent on the ability of the PSW System to cool the DGs. The long-term cooling capability of the RHR, core spray, and RHR service water pumps is also dependent on the cooling provided by the PSW System. In the analysis presented in [Edwin I. Hatch Nuclear Plant Updated Final Safety Analysis Report, Revision 38], only one PSW pump is required for safe shutdown, including RHR Shutdown Cooling System requirements.

The PSW subsystems are independent of each other to the degree that each has separate controls and power supplies, and the operation of one does not depend on the other. In the event of a DBA, one PSW pump is required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, two subsystems, each with two pumps, of PSW must be operable. At least one pump will operate if the worst single active failure occurs coincident with the loss of offsite power.

A subsystem is considered operable when it has an operable UHS, two operable pumps, and an operable flow path capable of taking suction from the intake structure and transferring the water to the appropriate equipment.

The 1A and 1C pumps form one PSW subsystem that provides cooling water to Division 1 essential equipment, including the 1A diesel generator. Similarly, the 1B and 1D pumps form the other PSW subsystem that provides cooling water to Division 2 essential equipment, including the 1C diesel generator. The 1B diesel generator can be aligned to a third bus in either unit and normally receives cooling water from the standby service water pump. The 1B diesel generator can also be aligned to receive cooling from the Hatch Unit 1 PSW system.

2.2 Regulatory Requirements and Guidance

Under 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit,” whenever a holder of a license wishes to amend the license, including TSs in the license, an application for amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate.

Under 10 CFR 50.36(c)(2), TSs must contain Limiting Conditions for Operation (LCOs), which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee must shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met. Typically, the TSs require restoration of equipment in a timeframe commensurate with its safety significance, along with other engineering considerations. Under 10 CFR 50.36(b), TSs must be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto.

In determining whether the proposed TS remedial actions should be granted, the Commission will apply the “reasonable assurance” standards of 10 CFR 50.40(a) and 50.57(a)(3). The regulation at 10 CFR 50.40(a) states that in determining whether to grant the licensing request, the Commission will be guided by, among other things, consideration about whether “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 any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in Part 20 of this chapter, and that the health and safety of the public will not be endangered.” The regulation at 10 CFR 50.57(a)(3) states that the Commission may issue an operating license amendment when it has, in part, reasonable assurance that the activities authorized by the operating license may be conducted without endangering the health and safety of the public.

Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (ADAMS Accession No. ML17317A256), describes a risk-informed approach to licensing basis changes that includes deterministic considerations to support this reasonable assurance finding. The guidance in RG 1.177, “Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications” (ADAMS Accession No. ML20164A034), utilizes the general guidance in RG 1.174 for application to changes in TS CTs and surveillance test intervals.

Guidance for the review of TSs is in Chapter 16.0, “Technical Specifications,” of NUREG-0800, Revision 3, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [Light-Water Reactor] Edition” (SRP), March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared Standard Technical Specifications (STS) for each of the LWR nuclear designs. Accordingly, the NRC staff’s review includes consideration of whether the proposed changes are consistent with the applicable reference STS, NUREG-1433, “Standard Technical Specifications [STS] General Electric BWR/4 Plants,” Volume 1 - Specifications, Revision 4.0 (ADAMS Accession No. ML12104A192).

Hatch, Unit 1, TS 1.3, "Completion Times" establishes the CT convention and provides guidance for its use. Hatch TS Limiting Condition for Operation (LCO) 3.0.1 through 3.0.8 contain usage requirements for LCOs. While not regulations, TS LCO 3.0.1 through 3.0.8 are license conditions and allow for evaluation of compliance with TS requirements.

3.0 TECHNICAL EVALUATION

3.1 Description of Current TS

When one PSW pump is inoperable, the current TS requires restoration of the inoperable pump to OPERABLE status within thirty days. If the pump is not restored in 30 days, the reactor must be in MODE 3 within the next twelve hours. MODE 3 is "Hot Shutdown," a condition where the reactor is shutdown, the coolant is above 212°F, and the core is adding decay heat.

The current TS 3.7.2 Condition A appears in the TSs as:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PSW pump inoperable.	A.1 Restore PSW pump to OPERABLE status	30 days

3.2 Description of Proposed TS

In the letter dated September 23, 2021, the licensee proposed adding REQUIRED ACTIONs A.2.1 and A.2.2 along with associated CTs and NOTES to TS 3.7.2 Condition A.

As proposed by the licensee, TS 3.7.2 Condition A would appear as:

A. One PSW pump inoperable.	A.1 Restore PSW pump to OPERABLE status	30 days
	OR -----------NOTES-------------------- 1. Only applicable during 1C PSW pump repair. 2. Only applicable until October 10, 2021 at 1620 EDT. --	
	A.2.1 Establish compensatory measures as described in letter NL-21-0862 dated September 23, 2021, Enclosure 5.	30 days
	AND	
	A.2.2 Restore PSW pump to OPERABLE status.	45 days

3.3 NRC Staff Evaluation

3.3.1 Risk-Informed Evaluation

RGs 1.174 and 1.177 describe an acceptable approach for developing risk-informed applications for proposed TS changes to CTs. All risk-informed applications for changes to plant TSs should explicitly address the five key principles stated in RG 1.174, Section C, and RG 1.177, Section C.2. These key principles are the following:

- Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption (i.e., a specific exemption under 10 CFR 50.12).

- Principle 2: The proposed licensing basis change is consistent with the defense-in-depth philosophy.

- Principle 3: The proposed licensing basis change maintains sufficient safety margins.

- Principle 4: When proposed licensing basis changes results in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants ("Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," 51 FR 30028 (Aug. 4, 1986)).

- Principle 5: The impact of the proposed licensing basis change should be monitored using performance measurement strategies.

Revision 3 of RG 1.174 identifies five key safety principles to be applied to risk-informed changes to the TSs. The NRC staff's evaluation of the licensee's proposed one-time extended CT against these key safety principles is discussed below.

3.3.1.1 Key Principle 1 (Meets regulations)

As stated in 10 CFR 50.36(c)(2):

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

When the necessary redundancy is not maintained (e.g., one Division of a two-Division system is inoperable), the TSs permit a limited period of time to restore the inoperable Division to operable status and/or take other remedial measures. If these actions are not completed within the specified CT, Hatch Unit 1 TS 3.7.2, Condition A, requires that the plant be placed in Mode 3. With one of two subsystems inoperable due to one inoperable pump, the safety function could be accomplished by the one operable pump in the inoperable subsystem or by either operable pump in the operable subsystem. The design of the Hatch Unit 1 PSW system allows for the inoperable subsystem to provide the required cooling to Division I, but the reliability is reduced because the redundant pump is not operable. This condition provides reasonable assurance that public health and safety would be protected because the period of time the condition may exist is limited by the one-time extension to the TS 3.7.2 CT for Condition A. Therefore, the proposed extension in CT satisfies current regulations.

3.3.1.2 Key Principle 2 (Defense in Depth)

Section 3.1 of the Enclosure to the letter dated September 21, 2021, SNC states, "During the extended Completion Time, the PSW System will remain within the limits of the Technical Specifications. Should an event occur requiring the PSW System and the UHS (i.e., the Altamaha River), the remaining PSW pumps are capable of performing the safety function of providing cooling water."

Defense-in-depth is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

As discussed throughout RG 1.174, consistency with the defense-in-depth philosophy is maintained by the following measures:

- Preserve a reasonable balance among the layers of defense.
- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
- Preserve adequate defense against potential common cause failures.
- Maintain multiple fission product barriers.
- Preserve sufficient defense against human errors.
- Continue to meet the intent of the plant's design criteria.

The licensee is proposing no changes to the design of the plant or any operating parameter, no new operating configurations, and no new changes to the design basis in the proposed changes to the TSs. Therefore, the proposed increase in completion time for restoration of the 1C PSW pump does not affect the balance among the layers of defense; the independence and diversity incorporated in the plant design, the fission product barriers, defense against human errors, or the intent of the plant's design criteria. The risk assessment directly considers the effect of the reduction in redundancy of the PSW pumps. However, the licensee is proposing an extended CT following a failure of one of four identically-designed PSW pumps and specific compensatory measures to manage the risk increase associated with the extended CT. Therefore, the potential for common cause failures should be assessed to ensure reasonable protections against common cause failures are present and the risk-management should not be overly reliant on the specified compensatory measures.

The NRC staff assessment of common cause failure potential is in Section 3.3.2 of this SE.

The NRC staff assessment of compensatory measures is in Section 3.3.3 of this SE.

The NRC staff has determined that SNC addressed Key Principle 2 for the one-time CT extension for the 1C PSW pump.

3.3.1.3 Key Principle 3 (Maintain Sufficient Safety Margin)

Section 2.2.2 of RG 1.177, Revision 2, states, in part, that sufficient safety margins are maintained when:

- Codes and standards or alternatives approved for use by the NRC are met.
- Safety analysis acceptance criteria in the FSAR are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties.

The licensee is not proposing in this application to change any quality standard, material, or operating specification. Acceptance criteria for operability of equipment are not changed and use of the extended CT only when the PSW system retains the capability to perform its safety function ensures that the current safety margins are retained.

The current license, which includes TSs, allows for one PSW pump to be out of service for 30 days, and was issued after the NRC determined there was reasonable assurance of public health and safety, and compliance with NRC regulations. The licensee is not proposing any changes to its design and will continue to meet applicable codes and standards. However, to maintain sufficient safety margin during the additional fifteen days that operation may continue while the PSW 1C pump is out of service, the licensee proposes using compensatory measures described in the LAR. Although the out of service condition of the 1C PSW pump eliminates redundancy in the subsystems providing cooling to Division I essential equipment, proposed compensatory measures reduce the potential for maintenance errors to challenge the proper functioning of the operable equipment. Therefore, the NRC finds that compensatory measures will maintain an acceptable safety margin during the extended CT.

3.3.1.4 Key Principle 4 (Meets Policy Statement)

The evaluation below addresses the NRC staff's philosophy of risk-informed decision making that when the proposed changes result in a change in core damage frequency (CDF) or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. The NRC staff evaluation of Key Principle 4 for the proposed TS change is described below.

As provided in Commission's Safety Goal Policy Statement, the qualitative safety goals are as follows:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

As further provided in Commission's Safety Goal Policy Statement, the following quantitative objectives are to be used in determining achievement of the above safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed change on plant operational risk. The Tier 1 review involves two aspects: (1) evaluation of the technical adequacy of the Probabilistic Risk Assessment (PRA) models and their application to the proposed change, and (2) evaluation of the PRA results and insights based on the licensee's proposed change.

Evaluation of PRA Acceptability

The licensee's PRA scope for this application includes internal events, internal flooding, fire, and seismic events during full power operation. SNC stated that an external event screening evaluation has eliminated all other hazard groups.

In RG 1.174, it states, in part that, "[t]he PRA analysis used to support an application is measured in terms of its appropriateness with respect to scope, level of detail, conformance with the technical elements, and plant representation. These aspects of the PRA are to be commensurate with its intended use and the role the PRA results play in the integrated decision process." The technical acceptability of the PRA must be compatible with the safety implications of the TS change being requested and the role that the PRA plays in justifying that change. That is, the more the potential change in risk and/or the greater uncertainty in that risk from the requested TS change, the more rigor that must go into ensuring the technical adequacy of the PRA. This applies to Tier 1, and it also applies to Tier 2 and Tier 3 to the extent that a PRA model is used.

In RG 1.200, Revision 2 (ADAMS Accession No. ML090410014), it describes one acceptable approach for determining whether the technical elements of the PRA, in total, or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decision making for LWRs. RG 1.200, Revision 2, endorses, with comments and qualifications, the use of: (1) the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications;" (2) Nuclear Energy Institute (NEI) 00-02, Revision 1, "Probabilistic Risk Assessment (PRA) Peer Review Process

Guidance" (ADAMS Accession No. ML061510619); and (3) NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard" (ADAMS Accession No. ML083430462). The ASME/ANS PRA standard provides technical supporting requirements in terms of three Capability Categories (CCs). The intent of the delineation of the Capability Categories within the supporting requirements is generally that the degree of scope and level of detail, the degree of plant specificity, and the degree of realism increase from CC I to CC III. In general, the NRC staff anticipates that current good practice (i.e., CC II of the ASME/ANS standard) is adequate for the majority of applications.

On May 3, 2017, the NRC staff transmitted its review results of Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations" (F&Os) (ADAMS Accession No. ML17079A427). The NRC accepted Appendix X for use by licensees to close F&Os that were generated during a peer review process.

The following sections describe the NRC staff's acceptance of the technical adequacy of Hatch's PRA model.

Internal Events and Internal Flooding PRA

The SNC completed a full scope peer review of the internal events and internal flooding PRA in November 2009 against RG 1.200, Revision 2, and associated PRA standard ASME/ANS PRA Standard RA-Sa-2009. Additionally, in July 2017, the licensee conducted an F&O closure in accordance with Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations" (F&Os). All but two findings associated with internal flooding were closed.

In response to PRA request for additional information (RAI) 3, regarding the details of any open F&Os and associated applicability to the results of this LAR, SNC clarified that two Internal Event F&Os remained open (F&O 1-9 and F&O 6-8).

The F&O 1-9 was written against SR AS-B3 and AS-C2, regarding missing discussion on the phenomenological conditions expected for each accident sequence related to Station Blackout (SBO) with usage of fire water. F&O 6-8 was written against SR HR-G6, regarding the Hatch Human Reliability Analysis document where the consistency check did not include comparison of human error probabilities (HEPs) in regard to scenarios context, plant history, procedures, operational practices, and experience. The licensee provided discussion on the approach it took to check for reasonableness. These F&Os do not directly impact the 1C PSW pump quantification. Additionally, the configuration risk profile at Hatch for the requested time has margin to the acceptance criteria describe in RG 1.177 for a one-time TS change. Therefore, the NRC staff did not evaluate the F&Os for technical acceptability.

As documented in the Hatch Diesel Generator Liner Replacement One-Time Technical Specification Completion Time Extension LAR (ADAMS Accession No. ML20213C715) and associated RAI responses (ML20236S786), the licensee conducted an additional focused-scope peer review for the internal flooding PRA using the guidance of NEI 05-04/07-12/12-06, and as a result the two open findings from the original peer review were closed.

Fire PRA

The SNC completed a full scope peer review of the fire PRA in June of 2016. Further, an F&O closure independent assessment was performed per Appendix X of NEI 05-04/07-12/12-06 in October 2017. All findings were closed per this review.

Prior to the licensee's review above, the NRC staff reviewed the quality of the Hatch PRA against the PRA standard ASME/ANS RA-Sa-2009 and RG 1.200 for transition to 10 CFR 50.48(c), National Fire Protection Association Standard (NFPA) 805. The NRC staff found the PRA quality was acceptable.

Seismic PRA

A peer review of the seismic PRA (SPRA) was completed in September 2016. The F&Os were closed using Appendix X of NEI 05-04/07-12/12-06. Two of the finding resolutions were considered a model upgrade and a subsequent focused- scope peer review was performed on those elements affected with no additional findings issued.

Prior to the licensee's review above, the NRC staff reviewed the quality of the Hatch SPRA against the PRA standard ASME/ANS RA-Sa-2009 and RG 1.200 for approval with the 10 CFR 50.69 program. The NRC staff found the SPRA quality was acceptable.

Plant Representation

In RG 1.174, it states that, "[t]he PRA results used to support an application are derived from a PRA model that represents the as-built and as-operated plant to the extent needed to support the application." That is, at the time of the application, the PRA should realistically reflect the risk associated with the plant.

The SNC stated in their LAR that the Hatch PRA maintenance and update processes and technical capability evaluations provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions.

Based on the description of the PRA model update process, the NRC staff concludes that the licensee's PRA maintenance and change process ensure that the PRA model would be updated as necessary to reflect the as-built and as-operated plant.

Technical Acceptability Conclusion

The NRC staff previously reviewed the technical acceptability of the Hatch internal events, internal flooding, fire, and seismic PRA against the PRA standard ASME/ANS RA-Sa-2009 and RG 1.200 in the NRC's licensing actions authorizing use of 10 CFR 50.48(c), NFPA 805 (ADAMS Accession No. ML20066F592), and 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors" (ADAMS Accession No. ML20077J704). The NRC staff found the PRA was sufficient to support those licensing actions. The previous findings provide evidence that the licensee's PRA can support this LAR for one-time CT extension.

Other External Hazards

The SNC performed a plant-specific evaluation of an extensive set of other external hazards (including high winds and external flooding). The results have been submitted previously to the NRC for the Hatch 50.69 LAR (ADAMS Accession No. ML18158A583) and subsequent RAI responses (ADAMS Accession No. ML19197A097). That evaluation has been performed by the licensee using the criteria in ASME PRA Standard RA-Sa-2009, Section 6. SNC concluded that all other external hazards can be screened from applicability at Hatch. Therefore, the licensee concluded that there is no apparent significant other external hazards risk contribution for this application. The NRC staff reviewed the information and concludes there is no significant external hazards.

PRA Modeling

This section addresses the requirements of Section 3.1 of RG 1.200, Revision 2, which directs the licensee to identify the portions of the PRA used in the LAR. Accordingly, the licensee used the Rev. 8 Phoenix One-Top Multi-Hazard Model (OTMHM) contained in SNC calculation H-RIE-PHOENIX-U01, Revision 3. This model has the required quantification and support files set up to calculate either zero-maintenance or average maintenance risks. It also implements several model enhancements identified during PHOENIX development and therefore represents the most accurate model of record available. In the LAR, the licensee clarified that maintenance unavailability events for equipment that was protected were left at their nominal values.

The Revision 8 Phoenix OTMHM model of record contains internal events, internal flooding, internal fire, and seismic hazards. All other hazards screened out as being very low risk. The model can be evaluated one hazard at a time or with all hazards activated. Each hazard model has been peer reviewed against the ASME/ANS Standard RA-Sa-2009, and all of the F&Os, except for two described in 3.7.1.2 above, have been closed. The licensee reviewed the quantification and uncertainty notebooks for each hazard model and did not find any assumption or uncertainty that would impact the results of this evaluation.

As described in RG 1.177, subsequent issues identified with the model would most likely impact the base and configuration specific models equally, therefore the staff concludes that delta risk calculations for a one-time TS change should not be impacted. The NRC staff finds that the licensee adequately described and justified the changes performed to the PRA model to support the risk analysis for the PSW pump 1C one-time CT extension.

PRA Results and Insights

The SNC provided the table below, to summarize its calculated incremental conditional core damage probability/incremental conditional large early release probability(ICCDP/ICLERP) for the proposed 15-day CT for the PSW pump 1C.

Input Parameter	**Value**
CDF_{BASE}	6.44×10^{-5} per year
CDF_{NEW}	8.25×10^{-5} per year
Delta CDF	1.80×10^{-5} per year
ICCDP for 45-day[1] LCO	**2.22×10^{-6}**
$LERF_{BASE}$	4.38×10^{-6} per year
$LERF_{NEW}$	4.75×10^{-6} per year
Delta LERF	3.76×10^{-7} per year
ICLERP for 45-day LCO	**4.63×10^{-8}**

The RG 1.177 provides sets of acceptance criteria for one-time CT extensions. The first is an ICCDP of less than 1.0×10^{-6} and an ICLERP of less than 1.0×10^{-7}. The second set of criteria is when the ICCDP is greater than 1×10^{-6} but less than 1×10^{-5}, and the ICLERP is greater

[1] The licensee is requesting an additional 15 days after expiration of the current 30-day LCO making it a total CT of 45 days.

than $1.0x10^{-7}$ but less than $1.0x10^{-6}$, in which case a licensee must implement effective compensatory measures to reduce the sources of increased risk.

For the PSW pump 1C outage, the licensee's calculated ICCDP meets the second set of criteria, so it is acceptable, but compensatory measures are therefore required.

In PRA RAI 4, the NRC staff requested that the licensee provide updated ICCDP/ICLERP estimates by running its fire PRA model with the NRC's adjusted common-cause failures (CCF) for PSW Pump fail to start (FTS) and fail to run (FTR) using the following values: Adjusted CCF FTS : 8.033 E-3 and Adjusted CCF FTR : 2.077 E-3. In response, the licensee provided updated Fire Core Damage Frequency (CDF) / Large Early Release Frequency (LERF), OTMHM CDF/LERF, and ICCDP/ICLERP estimates for both FTS and FTR cases due to CCF of PSW pumps. The NRC staff considered the most bounding quantification (e.g., double counting the FTS and FTR ICCDP/ICLERP estimates) and the licensee's results remained below the threshold of the acceptance criteria for ICCDP and ICLERP. Therefore, the NRC staff finds that the licensee's PRA results are acceptable for this application.

Tier 2: Avoidance of Risk-Significant Configurations

A licensee must provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out-of-service in accordance with the proposed TS change. The avoidance of risk-significant plant configurations limits potentially high-risk configurations that could exist if equipment, in addition to that associated with the proposed TS change, is simultaneously removed from service or other risk-significant operational factors such as concurrent system or equipment testing are involved. Therefore, Tier 2 helps ensure that appropriate restrictions are placed on dominant risk-significant configurations relevant to the proposed TS change.

The SNC performed a Tier 2 evaluation in the LAR. The licensee stated that the risk insights from this configuration were examined and it identified the necessary Compensatory Measures as listed in section 2.2. In response to PRA RAI 2, regarding the licensee approach to avoid any risk significant configurations, SNC described its approach to protecting equipment and ensuring its continued operation through regular operator rounds. Additionally, the licensee identified systems, structures, and components (SSCs) as a part of its Tier 2 evaluation, including the high pressure coolant injection (HPCI) system, which had an outage during the initial 30-day PSW Pump CT. The licensee provided additional HPCI performance information in its RAI response letter dated, September 23, 2021. The licensee described the circumstance surrounding the HPCI outage and confirmed that the issue was addressed. The NRC staff reviewed the remainder of the compensatory measures and finds the licensee Tier 2 evaluation to be acceptable.

Tier 3: Risk-Informed Configuration Risk Management

A Tier 3 program ensures that while a PSW pump is inoperable, additional activities will not be performed that could further degrade the capability of the plant to respond to adverse conditions, and as a result, increase plant risk beyond that assumed by the risk-informed licensing action. A Tier 3 program: (1) ensures that additional maintenance does not increase the likelihood of an initiating event intended to be mitigated by the out-of-service equipment such as redundant or associated systems or components, (2) evaluates the effects of additional equipment out-of-service during the PSW pump 1C maintenance activities that would adversely impact risk, and (3) evaluates the impact of maintenance on equipment or systems assumed to remain operable by the PSW pump CT analysis.

Accordingly, a licensee should develop a Configuration Risk Management Program (CRMP) to ensure that it appropriately evaluates the risk impact of out-of-service equipment before performing a maintenance activity. Licensees can utilize the overall CRMP (as referenced in RG 1.177) through the Maintenance Rule (10 CFR 50.65(a)(4)). Specifically, the rule requires that, before performing any maintenance activity, the licensee must assess and manage the potential risk increase that may result from a proposed maintenance activity. A licensee's submittal must include a discussion of the licensee's CRMP for assessing the risk associated with the removal of a PSW pump from service and its conformance to the requirements of 10 CFR 50.65(a)(4), and the additions and clarifications outlined in Section 2.3.7.2 of RG 1.177, as they relate to the proposed extended PSW pump CT.

In Section 2.2 of the letter dated September 21, 2021, the licensee stated that Hatch has an on-line configuration risk management process. SNC also stated that the CRMP uses the same hazard models that were used for their evaluation, and since the process evaluates planned work as well as current configurations, it will identify any potential high-risk conditions during the extended CT, including any rigging and lifting of the PSW pump and motor near other components that remain in service.

Based on the above, the NRC staff finds the licensee's Tier 3 program is consistent with RG 1.177 and, therefore, is acceptable.

Key Principle 4 Conclusion

The NRC staff finds the licensee has demonstrated that the scope, level of detail, and technical adequacy of its PRA models are sufficient to support the proposed one-time CT change. The risk metrics used to support the LAR are consistent with RG 1.177. The NRC staff further finds that the licensee has followed the three-tiered approach outlined in RG 1.177 to evaluate the risk associated with the proposed change and, therefore, the proposed change satisfies the fourth key safety principle of RG 1.174.

3.3.1.5 Key Principle 5 (Monitor Impact)

Guidance in RG 1.177 establishes the need for an implementation and monitoring program to ensure that extensions to TS CTs would not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms.

An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. RG 1.174 states that monitoring performed in conformance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application.

In its LAR, the licensee stated that the impact of the proposed change will be monitored for effectiveness in accordance with the existing plant maintenance rule program pursuant to 10 CFR 50.65(a)(4) and the associated implementation guidance in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (ADAMS Accession No. ML18220B281). In response to RAI 7 dated September 23, 2021, SNC stated PSW pumps are monitored by the condition-based monitoring and in-service testing programs.

Based on the above, the NRC staff concludes that the implementation and monitoring program for the proposed one-time 1C PSW ump extension satisfies the fifth key safety principle of RG 1.177.

Risk Evaluation Summary

With respect to the risk due to the one-time extension of the CT of TS 3.7.2 from 30 days to 45 days, the NRC staff finds the proposed changes meet the following five key principles:

1. The proposed change meets the current regulations and applicable order based on deterministic evaluations.

2. The proposed change is consistent with the defense-in-depth philosophy by the compensatory measures and backup equipment, including FLEX.

3. The proposed change maintains sufficient safety margins.

4. The risk impact of the licensee's request as estimated by ICCDP and ICLERP is consistent with the acceptance guidelines specified in RG 1.177 and the NRC staff guidance outlined in Sections 19.1 and 16.1 of NUREG-0800 (Standard Review Plan). The licensee's methodology for assessing the risk impact is accomplished using PRA models of sufficient scope and technical adequacy. The licensee has followed the three-tiered approach and performance monitoring programs outlined in RG 1.177.

5. The impact of the proposed changes is being monitored using performance measurement strategies under the plant's Maintenance Rule program.

Therefore, the NRC staff finds the proposed one-time CT extension for TS 3.7.2 will have minimal impact on the continued safe operation of the plant, and are, therefore, acceptable.

3.3.2 Common Cause Failure Assessment

The NRC staff evaluated the potential for common cause failures considering the issues that resulted in the continued inoperability of the 1C PSW pump. SNC initially discovered the pump to be inoperable through normal operator rounds on August 26, 2021, when an operator identified excessive vibration. The licensee shut down the pump and identified the following conditions through troubleshooting:

- All four motor to pump discharge fasteners were loose and could be turned by hand
- One of the pump discharge head to floor fasteners was loose
- A significant gap existed between the seal box drive collar and gland plate assembly
- The suction head was no longer connected to the pump column

The SNC stated that investigation into the cause of the initial failure of PSW pump 1C was in progress, but the licensee described that a leading theory involved fatigue-induced failure of the pump shaft that led to excessive vibration in the pump and motor.

In evaluating the potential for common cause failures, the NRC staff considered evidence related to potential causes of the initial pump failure, maintenance history, and performance monitoring of the PSW pumps. The licensee stated that likely conditions leading to fatigue failure of the pump shaft include (1) an internal flaw in the pump shaft, (2) misalignment of the pump and motor during motor replacement activity in January 2021, or (3) age related degradation of the components securing the pump shaft. SNC provided an assessment of each of these conditions in the supplement to the LAR. The licensee also described operating experience with deep-draft pumps in the SNC operating fleet.

The maintenance history of the pumps is relevant to each of the conditions potentially leading to shaft failure. The licensee stated that the 1C PSW pump was last replaced in 2013, 1A in 2017, 1B in 2021, and 1D in 2018. SNC also stated that replacement of the pump included replacement of the pump shaft, which are procured to a specific standard and have a unique certificate of conformance. Therefore, the licensee concluded that the potential for an internal flaw in the pump shaft to result in common failure of a second pump was considered remote based on the several years separating the 2013 replacement of the 1C pump and the other pump replacements beginning in 2017. SNC also considered misalignment of the pump and motor to be unlikely to result in common failure of a second pump. The 1C pump motor was replaced in January 2021, and was the first pump to undergo a new maintenance practice to verify pump-motor alignment. The 1B pump also underwent this alignment process in conjunction with pump replacement in May 2021, and no adverse conditions have been identified. The licensee concluded that condition monitoring, including pump vibration measurements within specified limits, provide reasonable assurance that misalignment is not a likely contributor to near-term failure of the 1B PSW pump, and the staff agreed with this assessment. The remaining pumps have not been subject to major repairs since 2018. The potential for wear or age-related degradation of components was most likely to affect the 1C PSW pump because it had by far the greatest run time since last replacement of the PSW pumps.

The licensee reviewed SNC operating experience related to deep-draft pumps similar to the Hatch PSW pumps. The operating experience included an example of mis-alignment due to positioning of seismic restraints on the pump column, but this condition did not exist for the 1C PSW pump. No other operating experience related to the potential causes of the initial failure was identified.

The SNC described performance monitoring of the PSW pumps. The 1C PSW pump review back to 2016 indicated a consistently declining trend in differential pressure that reached the Alert Range of the American Society of Mechanical Engineers (ASME) in-service testing program during testing on June 29, 2021. However, none of the monitored parameters indicated unacceptable performance of the 1C PSW pump. The 1A PSW pump entered the Required Action Range for differential pressure in November 2020 and the Alert Range for vibration as of May 2021. The licensee stated that the 1A PSW pump has been scheduled for replacement in November 2021, and the performance monitoring data supports continued operation beyond that date. The licensee stated that the remaining PSW pumps have been continuously operated and performance monitoring indicated no degradation. Therefore, the licensee considered the 1A, 1B, and 1D PSW pumps operable with a low probability of failure during the extended CT for restoration of the 1C PSW pump.

The NRC staff assessed the information provided by SNC regarding the potential causes of the failure of the 1C PSW pump and the likelihood that those potential causes could affect the remaining operating PSW pumps. The NRC staff determined that the difference in operating time since the last pump replacement between the 1C PSW pump and the other PSW pumps reduces the likelihood that material defects or aging effects would reasonably result in a common cause failure of any of the remaining operating pumps over the next few months of operation. Although the recent replacement of the 1B PSW pump could indicate the potential for misalignment to result in premature failure, performance monitoring of this pump does not indicate abnormal vibration that would be associated with significant misalignment.

Performance monitoring does indicate some degradation of the 1A PSW pump, but the levels do not indicate the need for immediate maintenance and are consistent with the operating time of the pump. Therefore, the staff concludes the potential for common cause failure is not elevated and the modeling of increased common cause factors associated with failure of the 1C PSW pump in the risk assessment adequately captures the common cause failure risk.

3.3.3 Compensatory Measures

In Attachment 5 to the LAR supplement dated September 23, 2021, SNC presented a list of compensatory measures proposed to be implemented during the extended completion time period. The licensee listed the following compensatory measures:

- The following equipment is protected as required by SNC Procedure NMP-OS-010-002 (Reference 3) for 1C PSW pump out-of-service:
    - 1A PSW Pump
    - 1E 4160V Frame 3 (power supply to 1A PSW Pump)
    - 1A PSW Pump Control Switch
- Travelling water screen 1B will be placed in RUN if the 1A screen is taken out of service.
- HNP Operations (each shift) will review the abnormal procedure for loss of PSW, SNC Procedure 34AB-P41-001-01 (Reference 4).
- PSW Pumps 1A, 1B, and 1D will be protected with work limited to TS required surveillances only.
- High Pressure Coolant Injection (HPCI) will be protected with work limited to TS required surveillances only.
- Reactor Core Isolation Cooling (RCIC) will be protected with work limited to TS required surveillances only.
- No maintenance will be performed on 1T48F081 or 1T48F082, the Containment Hardened Vent path.
- The 1B diesel generator and the Standby Service Water (SSW) pump will be protected, and work limited to TS required surveillances only.
- All three Unit 1 startup transformers and their associated 230KV breakers will be protected.
- No preventive maintenance will be performed on the FLEX pumps to ensure their availability during the extended Completion Time.

The NRC staff reviewed the proposed compensatory measures with respect to reducing the risk associated with certain initiating events related to loss of off-site power or loss of the normal heat sink to verify these administrative measures were not overly relied upon to manage risk. The NRC staff reviewed the information and concluded that the measures to protect equipment are appropriate and conservative with respect to the risk assessment in that the measures reduce the likelihood of maintenance affecting the availability of risk-important components. The licensee did not overly rely on temporary or portable equipment to reduce risk because the only such equipment considered within the scope of the compensatory measures was the FLEX equipment pumps and generators. The licensee stated that this equipment was modeled in the seismic risk assessment, but not for other potential initiators. Therefore, the NRC staff concluded that the reliance on these measures to reduce risk is appropriate and not inconsistent with the risk assessment modeling.

3.3.4 TS Evaluation

The NRC staff reviewed the proposed changes to TS 3.7.2 Condition A. SNC provided an evaluation to justify the proposed changes. As discussed in the sections above, the NRC staff determined the evaluation adequately justifies the allowance to maintain the reactor in MODE 1 with one inoperable PSW pump 15 days beyond the current 30 CT. Therefore, the NRC staff determined the TS will continue meet the requirements of 10 CFR 50.36(b), because the TS will continue to be based on the analyses and evaluation included in the safety analysis report, and amendments thereto.

The NRC staff's review included an evaluation of the proposed TS changes for conformance to the CT conventions in Hatch TS 1.3 and alignment with requirements contained in Hatch TS LCOs 3.0.1 through 3.0.8 to ensure the proposed change, once implemented, will continue to provide reasonable assurance of adequate protection of public health and safety. As part of its evaluation, the NRC staff considered how the changes impact implementation of existing requirements within TS 3.7.2. On pages E1-8 through E1-9 of the letter dated September 23, 2021, SNC provided an explanation of how the changes would be implemented. The NRC staff notes that these actions and CTs are only applicable to the 1C PSW pump repair until 4:20 EDT, October 10, 2021. The NRC staff further noted the allowance is only applicable if the Compensatory Measures listed in Enclosure 5 of the licensee's letter dated September 23, 2021, are implemented as described in the letter.

In its letter dated September 23, 2021, the licensee stated:

> If during the extended Completion Time it is discovered that any of the compensatory measures are found to be not implemented (i.e., Required Action A.2.1 not met), then the plant would be in Condition E, which would require Unit 1 to be in Mode 3 in 12 hours. If the Compensatory Measures can be restored while the plant is in Condition E, then Condition E can be exited, and the plant would resume under Required Actions A.2.1 and A.2.2. The Condition A Completion Time would not reset, but would continue from the time the 1C PSW pump was first declared inoperable (specifically the 45 day "clock" would continue from 1620 EDT, August 26, 2021). This concept is discussed in Example 1.3-2 of Plant Hatch Technical Specification 1.3, Completion Times.

The NRC staff determined the proposed change conforms to the CT conventions in Hatch TS 1.3 and aligns with requirements in the Hatch TS LCOs 3.0.1 through 3.0.8. The NRC staff determined that while the proposed additions to the TS represent a variation from the STS, the existing logic structure and usage rules in the Hatch TS can be used to determine proper implementation of the change and allow for continued evaluation of compliance with TS requirements during the additional fifteen day period.

The NRC staff determined the proposed change allows the licensee to keep the reactor in MODE 1 with one PSW pump inoperable an additional fifteen days compared to the current requirement. The NRC staff determined the allowance consists of appropriate remedial actions SNC can take when LCO 3.7.2 is not met. Therefore, the NRC staff determined that the TS, as amended by the proposed change, will continue to meet the regulatory requirements of 10 CFR 50.36(c)(2).

The addition of the NOTES and REQUIRED ACTIONs A.2.1 and A.2.2 to TS 3.7.2 resulted in information being moved to the next page. The NRC staff reviewed the change, and the NRC staff finds the movement of information to the next page necessary and acceptable.

NRC Staff Conclusion

TS LCO 3.7.2 requires the two PSW subsystems to be operable. SNC proposed changes to allow continued operation in MODE1 for fifteen additional days beyond the current thirty day limit with one PSW pump inoperable. The NRC staff concludes, based on information provided by the licensee, that the changes in risk associated with extending the associated CT are less than that of the guidance thresholds in RG 1.177 and RG 1.174, and, therefore, support the one-time extension of the CT associated with the inoperability of the 1C PSW pump. Therefore, the NRC staff concludes that the licensee's proposed changes to TS 3.7.2 are acceptable, because the changes meet regulatory requirements and provide reasonable assurance of adequate protection of public health and safety.

4.0 EMERGENCY SITUATION

Background

The NRC's regulations in 10 CFR 50.91(a)(5) state that where the NRC finds that an emergency situation exists, in that failure to act in a timely way would result in derating or shutdown of a nuclear power plant, or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. In such a situation, the NRC will publish a notice of issuance under 10 CFR 2.106, providing for opportunity for a hearing and for public comment after issuance.

As discussed in SNC's application dated September 21, 2021, the licensee requested that the proposed amendments be reviewed by the NRC on an emergency basis. The licensee stated that the emergency situation resulted from the unforeseen failure of the Hatch, Unit 1, "C" PSW pump and motor during post-maintenance testing (PMT).

SNC stated it has taken the following actions since August 26, 2021.

- On August 26, abnormal noise and observed vibration of the motor. SNC removed the pump from service and declared it INOPERABLE (30-day LCO).

- On August 30, uncoupled run was completed with no issue; motor was operating as expected. The 1C PSW pump replacement was scheduled to be completed on September 4, 2021.

- On August 31, SNC investigation found several motor hold-down bolts loose (could be turned by hand). While removing the 1C PSW pump, the pump shaft was discovered to be broken inside the coupling, where the bottom pump column bolts to the pump. The bolts at the pump-to-pump column flanged connection were also discovered to be loosened/broken. The 3-stage pump itself remained in the intake suction pit. In addition, 4 of the 16 bolts were discovered to be missing, and 5 bolts were partially broken. All 16 nuts remained in the intake structure. Plans were developed by SNC to retrieve the existing 1C PSW pump and parts from the suction pit. SNC scheduled divers to retrieve the pump. The 1C PSW pump replacement was scheduled to complete by September 6, 2021.

- On September 7, the pump was recovered by SNC and the pump alignment was in progress. SNC determined that a seismic restraint required modification.

- On September 13, SNC completed the modification to the seismic restraint. The 1C PSW pump was installed, aligned, and SNC cleaned the intake.

- On September 14, SNC completed cleaning of the intake.

- On September 16, when the 1C PSW pump was started after maintenance, there was a high-pitched noise during startup that did not subside, and the pump was secured.

- On September 17, SNC identified that the lower guide bearing caps were axially misaligned by 0.030 inch, which led to heavy wear, or wiping, of the lower guide bearing during the PMT.

- On September 18, SNC identified that the thrust bearing had been dislodged from its housing.

As a result of these circumstances, the licensee cannot complete the installation of the 1C PSW pump and motor by the 30-day CT and requested an additional 15 days to complete the installation and return the pump to OPERABLE.

Based on its discovery on September 18, that the 1C PSW pump and motor needed to be replaced, SNC stated that neither a routine nor an exigent amendment could be processed prior to September 25, 2021.

NRC Staff Conclusion

The NRC staff reviewed the licensee's basis for processing the proposed amendment as an emergency amendment and has determined that an emergency situation exists consistent with the provisions in 10 CFR 50.91(a)(5). Furthermore, the NRC staff determined that: (1) the licensee used its best efforts to make a timely application; (2) the licensee could not reasonably have avoided the situation; and (3) the licensee has not abused the provisions of 10 CFR 50.91(a)(5). Based on these findings, and the determination that the amendment involve no significant hazards consideration as discussed below, the NRC staff has determined that a valid need exists for issuance of the license amendments using the emergency provisions of 10 CFR 50.91(a)(5).

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee's evaluation of the issue of no significant hazards consideration is presented below:

1. Does the proposed amendment [change] involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change involves a one-time extension to the Completion Time for TS 3.7.2 Condition A to allow necessary time to restore the 1C PSW pump to operable status. The proposed amendment does not affect accident initiators or precursors nor adversely alter the design assumptions, conditions, and configuration of the facility. The proposed amendment does not alter any plant equipment or operating practices with respect to such initiators or precursors in a manner that the probability of an accident is increased. The proposed amendment will not alter assumptions relative to the mitigation of an accident or transient event. Furthermore, the PSW System will remain capable of adequately responding to a design basis event or transient during the period of the extended Completion Time.

   Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment [change] create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed amendment does not introduce any new or unanalyzed modes of operation. The proposed changes do not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis.

   Therefore, the proposed amendment will not create the possibility of a new or different accident from any accident previously evaluated.

3. Does the proposed amendment [change] involve a significant reduction in a margin of safety?

   Response: No.

   The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment. The performance of these fission product barriers is not affected by the proposed amendment; therefore, the margins to the onsite and offsite radiological dose limits are not significantly reduced.

In addition, during the extended Completion Time, the PSW System will remain capable of providing the required cooling to systems responsible for mitigating the consequences of a design basis event such as a LOCA.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above evaluation, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendments and that the amendments should be issued as allowed by the criteria contained in 10 CFR 50.91.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified on September 21, 2021, and the State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated; or (b) create the possibility of a new or different kind of accident from any accident previously evaluated; or (c) involve a significant reduction in a margin of safety; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (3) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: September 24, 2021

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATION 3.7.2, “PLANT SERVICE WATER (PSW) SYSTEM AND ULTIMATE HEAT SINK (UHS)” (EPID L-2021-LLA-0164) (**EMERGENCY CIRCUMSTANCES**) DATED SEPTEMBER 24, 2021

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