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

April 22, 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. 2 - ISSUANCE OF AMENDMENT NO. 254, REGARDING REVISION TO TECHNICAL SPECIFICATIONS FOR ONE-TIME EXTENSION OF COMPLETION TIMES RELATED TO RESIDUAL HEAT REMOVAL SYSTEM **(EMERGENCY CIRCUMSTANCES)** (EPID L-2021-LLA-0068)

Dear Ms. Gayheart:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 254 to Renewed Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit No. 2. The amendment consists of changes to the technical specifications (TSs) in response to your application dated April 19, 2021, as supplemented on April 20 (two letters), and April 22, 2021.

The amendment makes a revision to Required Action A.2 of TS 3.5.1, “ECCS [Emergency Core Cooling System] – Operating,” to extend the Completion Time (CT) from 7 days to 15 days only while repairs of the 2D Residual Heat Removal (RHR) pump repair are ongoing, and only until May 1, 2021. This amendment allows Southern Nuclear Operating Company, Inc. (SNC) to continue to operate while it performs repair and testing activities on the 2D RHR pump, which has been inoperable since April 16, 2021.

The license amendment is issued under emergency circumstances as provided 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. In this instance, an emergency situation exists in that the proposed amendment is needed to allow the licensee to preclude a plant shutdown.

A copy of the related safety evaluation (SE) is also enclosed. The SE 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 next monthly *Federal Register* notice.

If you have questions, you can contact me by phone at 301-415-3100 or by email 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. 50-366

Enclosures:
1. Amendment No. 254 to NPF-5
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. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 254
Renewed License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 2 (the facility) Renewed Facility Operating License No. NPF-5 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 April 19, 2021, as supplemented on April 20 (two letters), and April 22, 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 set forth in 10 CFR Chapter I;

   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 to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-5 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. 254 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

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2021.04.22 19:46:21 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. NPF-5 and Technical Specifications

Date of Issuance: April 22, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 254

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

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
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3
3.5-4	3.5-4

(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[2] 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, in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 254, 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) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

(a) 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

2 The original licensee authorized to possess, use, and operate the facility was Georgia Power Company (GPC). Consequently, certain historical references to GPC remain in certain license conditions.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six of seven safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One LPCI pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days
	OR -----NOTES----- 1. Only applicable during 2D RHR pump repair. 2. Only applicable until May 1, 2021. ----------	
	A.2.1 Establish compensatory measures as described in letter NL-21-0423, dated April 22, 2021. AND	7 days
	A.2.2 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	15 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	12 hours
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	1 hour 14 days
D. HPCI System inoperable. AND Condition A entered.	D.1 Restore HPCI System to OPERABLE status. OR D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
E. Required Action and associated Completion Time of Condition 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. Two or more ADS valves inoperable.	F.1 Be in MODE 3. AND F.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. <u>OR</u> HPCI System and two or more ADS valves inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.2	---------------------------NOTES---------------------------- 1. Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the Residual Heat Removal (RHR) low pressure permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. 2. Not required to be met for system vent flowpaths opened under administrative control. --- Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.3	Verify ADS air supply header pressure is ≥ 90 psig.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.4	Verify the RHR System cross tie valve is closed and power is removed from the valve operator.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.5	(Not used.)	
SR 3.5.1.6	------------------------------NOTE---------------------------- Only required to be performed prior to entering MODE 2 from MODE 3 or 4, when in MODE 4 > 48 hours. --- Verify each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position.	In accordance with the Surveillance Frequency Control Program
SR 3.5.1.7	Verify the following ECCS pumps develop the specified flow rate against a system head corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF CS / ≥ 4250 gpm / 1 / ≥ 113 psig LPCI / ≥ 17,000 gpm / 2 / ≥ 20 psig	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.1.8	--------------------------NOTE----------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1058 psig and ≥ 920 psig, the HPCI pump can develop a flow rate ≥ 4250 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program

(continued)

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 254 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-366

1.0 INTRODUCTION

By application dated April 19, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21109A388), as supplemented on April 20 (two letters), and April 22, 2021 (ADAMS Accession Nos. ML21110A652, ML21110A722, and ML21112A067, respectively), Southern Nuclear Operating Company, Inc. (SNC, the licensee), requested changes to the technical specifications (TSs) for the Edwin I. Hatch Nuclear Plant, Unit No. 2 (Hatch).

The amendment would add Required Action A.2 of TS 3.5.1, "ECCS [Emergency Core Cooling System] – Operating," and make a corresponding new Completion Time (CT) from 7 days to 15 days. This amendment allows SNC to continue to operate while SNC performs repair and testing activities on the 2D Residual Heat Removal (RHR) pump.

The licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff process this submittal as an emergency amendment.

2.0 REGULATORY EVALUATION

2.1 System Description

The purpose of the ECCS at Hatch, Unit 2, is to mitigate design-basis loss-of-coolant accidents (LOCAs) and satisfy the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46 for ECCS performance criteria.

The RHR system includes a number of pumps and heat exchangers that can be used to cool the nuclear steam supply system under a variety of situations. During normal shutdown and reactor servicing, the RHR system removes residual and decay heat. The RHR system allows decay heat to be removed whenever the main heat sink (main condenser) is not available. Another operational mode of the RHR system is low pressure coolant injection (LPCI). The LPCI operation is an engineered safety feature system for use during a LOCA.

Enclosure 3

In the submittal dated April 19, 2021, SNC stated:

> Note that while the RHR System functions in several modes of operation ..., SNC has determined that only the low pressure coolant injection (LPCI) mode, as addressed by TS 3.5.1, is adversely affected with a single RHR pump in a loop Inoperable. Other RHR system modes such as Suppression Pool Cooling (TS 3.6.2.3), Suppression Pool Spray (TS 3.6.2.4), and Drywell Spray (TS 3.6.2.5) only require a single RHR pump in a loop to be Operable and are thus not affected by this request.

2.2 Licensee Proposed TS Changes

Under 10 CFR 50.36(c)(2)(i), when a limiting condition for operation (LCO) of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. The current TSs' remedial actions for when the LCO for the ECCS system is not met do not provide the necessary time to repair the pump that is inoperable. Consequently, the additional time needed to repair the pump is outside of the CT allowed by the TSs and would require the licensee to shut down the reactor. The proposed change would add remedial actions to address the circumstance.

TS LCO 3.5.1 "ECCS - Operating," requires that each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six of seven safety/relief valves shall be OPERABLE in MODES 1, and in MODES 2 and 3 (except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure less than 150 psig.) With one low pressure ECCS injection/spray subsystem or one LPCI pump in both LPCI subsystems inoperable, then the licensee must restore the low pressure subsystem(s) to operable within seven days. The proposed amendment would provide TS 3.5.1 with additional Required Actions A.2.1 and A.2.2 to address an ongoing Hatch, Unit 2, 2D RHR pump repair. The new actions and CTs provide the licensee an additional 8 days (15 days from the time of discovery) to repair the 2D RHR pump and is specifically related to the first component of TS 3.5.1 Condition A, "One low pressure ECCS injection/spray subsystem inoperable." TS 3.5.1 Condition B, which addresses what must be done if the Required Action and associated CT of Condition A are not met, would continue to require the licensee to place the reactor in MODE 3 (Hot Shutdown) within 12 hours.

The additional Required Actions, CTs, and Notes are as follows:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. <u>OR</u> One LPCI pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status. *<u>OR</u>* ----------------*NOTES*----------------- *1. Only applicable during 2D RHR pump repair.*	7 days

	2. Only applicable until May 1, 2021.	
A.2.1	*Establish compensatory measures as described in letter NL-21-0423, dated April 22, 2021.*	*7 days*
	AND	
A.2.2	*Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.*	*15 days*

Furthermore, the proposed addition of the Notes and Required Actions A.2.1 and A.2.2 would cause Condition B to move to the next page. The proposed movement of Condition B would cause pages 3.5-2 through 3.5-4 to have editorial changes; there are no proposed technical changes. So, the proposed changes to pages 3.5-2 through 3.5-4 are editorial in nature.

2.3 Regulatory Requirements

Under 10 CFR 50.90, whenever a holder of a license wishes to amend the license, including TSs in the license, an application for an amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a), in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate. Under the common standards for licenses and construction permits in 10 CFR 50.40(a), the Commission will be guided by, among other things, whether the operating procedures, the facility and equipment, the use of the facility, and other TSs, collectively provide reasonable assurance that the applicant will comply with the regulations and that the health and safety of the public will not be endangered. Additionally, the considerations specifically for issuance of operating licenses in 10 CFR 50.57(a)(3) similarly provide that there must be reasonable assurance that the activities at issue will not endanger the health and safety of the public.

The regulation in 10 CFR 50.36(c)(2)(i) states, in part, that:

> 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.

The remedial actions must provide the requisite reasonable assurance of public health and safety.

Licensees may propose revisions to the TSs. The NRC staff reviews proposed changes and will generally issue changes provided that the plant-specific review supports a finding of continued adequate protection of public health and safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered), (2) the change is more restrictive than the licensee’s current requirement, or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework, and additional specialized guidance, is discussed in Section 3.0 of this safety evaluation in the context of the proposed TS changes contained in the licensee's license amendment request (LAR).

The regulations in 10 CFR 50.46(b) establish acceptance criteria for ECCS evaluations for light-water nuclear power reactors, as summarized below:

- Peak cladding temperature- the calculated maximum fuel element cladding temperature shall not exceed 2,200 degrees Fahrenheit (°F).

- Maximum cladding oxidation - the calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

- Maximum hydrogen generation - the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

- Coolable geometry - calculated changes in core geometry shall be such that the core remains amenable to cooling.

- Long-term cooling - after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The applicable 10 CFR Part 50, Appendix A, "General Design Criteria [Criterion] for Nuclear Power Plants," was considered as follows:

- Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

  Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

- Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

- Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

2.4 Regulatory Guidance

The regulatory guidance that the NRC staff used in its review of the risk information submitted in support of the LAR consisted of Regulatory Guide (RG) 1.174, RG 1.177, and RG 1.200 as described below:

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment [PRA] in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 (ADAMS Accession No. ML17317A256), describes an acceptable method for licensees and the NRC to use for assessing the nature and impact of proposed changes to the licensing basis by considering engineering issues and applying risk insights. This regulatory guide also provides risk-acceptance guidelines for evaluating the results of such evaluations.

- RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 1, May 2011 (ADAMS Accession No. ML100910008), describes methods acceptable to the NRC for assessing the nature and impact of proposed permanent TS changes, including allowed outage times, by considering engineering issues and applying risk insights. This regulatory guide also provides risk-acceptance guidelines for evaluating the results of such assessments.

- RG 1.200, "Acceptability of Probabilistic Risk Assessment Results of Risk-Informed Activities," Revision 2, March 2009 (ADAMS Accession No. ML090410014), describes one approach acceptable to the NRC staff for determining whether a base PRA, in total or in the portions that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors.

In addition to risk information, the NRC staff used guidance for the review of TSs 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 (i.e., the current STS), as modified by NRC-approved travelers. The NRC used "Standard Technical Specifications, General Electric BWR/4 Plants," NUREG-1433, Volume 1, "Specifications," and Volume 2, "Bases," Revision 4.0, April 2012 (ADAMS Accession Nos. ML12104A192 and ML12104A193, respectively).

3.0 TECHNICAL EVALUATION

3.1 Risk-Informed Evaluation

An acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary changes, is to show that the proposed changes meet the five key principles stated in RG 1.174, Section 2, and RG 1.177, Section B. These key principles are

- Principle 1: The proposed licensing basis change meets the current regulations unless itis 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 defense-in-depth [DID] philosophy.

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

- Principle 4: When proposed licensing basis changes result 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 cited above.

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

3.2 Key Principle 1 (Meets regulations)

10 CFR 50.36(c) requires technical specifications to include certain categories of items, including limiting conditions for operations. The licensee's proposed change does not alter the categories in the licensee's TS, and does not alter compliance with 10 CFR 50.36(c). Further, as discussed in the sections below, the NRC staff finds that the licensee's proposal meets the requisite reasonable assurance regulations.

3.3 Key Principle 2 (Defense in Depth)

Section 3.2 of the Enclosure to the LAR dated April 19, 2021, states "The ECCS system has sufficient capacity to function for design basis events while in Condition A. Assuming no additional failures, the [Updated Final Safety Analysis Report] UFSAR acceptance criteria for the design events will be met should such an event occur during the time that the 2D RHR pump is out of service."

The NRC staff reviewed the Hatch UFSAR. The NRC staff notes that, as discussed in Section 6.3.3.3 and Tables 6.3-2 and 6.3-4 of the UFSAR, dated September 29, 2020 (ADAMS Accession No. ML20303A186), the limiting single failure considered for LOCA analysis is a dedicated diesel C battery failure which results in the loss of two of the four LPCI pumps. Because the UFSAR shows that the applicable requirements and criteria can be met with two LPCI pumps out of service, the NRC staff concluded that with only one pump out of service, the acceptance criteria will likewise continue to be met. This finding is consistent with the existing TS, which already allow continued operation for a week while a single pump is out of service.

However, in light of the additional week of outage time being requested, the licensee proposed to implement additional compensatory measures and risk management controls. These new actions and controls address various plant maintenance configurations to maintain and manage acceptable risk levels during maintenance evolutions for 2D RHR pump. The intent of the compensatory measures is to reduce the duration of risk-sensitive activities and avoid high-risk sensitive equipment outages or maintenance states that result in high-risk plant configurations.

The NRC staff concludes these steps are consistent with the DID philosophy.

3.4 Key Principle 3 (Maintain Sufficient Safety Margin)

The existing license, which includes TS, allows for one RHR pump to be out of service for a week, and was issued after the NRC determined there was reasonable assurance of public health and safety, and of compliance with the NRC's regulations. The licensee is not proposing any changes to its design. However, to maintain sufficient safety margin during the additional eight days that operation may continue while the RHR pump is out of service, the licensee proposes using compensatory measures described in the licensee's letter NL-21-0423, dated April 22, 2021. The NRC finds that those compensatory measures will maintain a safety margin during the extended AOT duration.

3.5 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 significantadditional 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 competingtechnologies 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 towhich 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.

3.5.1 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 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.

3.5.1.1 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. The licensee stated that an external event screening evaluation has eliminated all other hazard groups.

RG 1.174 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 or the greater uncertainty in that risk from the requested TS change, or both, 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.

RG 1.200, Revision 2, 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 decisionmaking 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 3.5.1.2 to 3.5.1.7 describe the NRC staff's acceptance of the technical adequacy of Hatch's PRA model.

3.5.1.2 Internal Events and Internal Flooding PRA

The licensee 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.

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.

3.5.1.3 Fire PRA

The licensee 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 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 found the quality was acceptable.

3.5.1.4 Seismic PRA

A peer review of the seismic PRA 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.

For this application, the licensee asserted that the seismic results were not sensitive to maintenance on the RHR pump and therefore the quantified results are not used. The NRC staff found that approach to be reasonable.

3.5.1.5 Plant Representation

RG 1.174 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 licensee stated in the 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.

3.5.1.6 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), National Fire Protection Association Standard (NFPA) 805 (ADAMS Accession Nos. 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 2D RHR pump out-of-service time extension.

3.5.1.7 Other External Hazards

The licensee 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 license amendment request (LAR) (ADAMS Accession Number ML18158A583) and subsequent RAI responses (ML19197A097). That evaluation has been performed by the licensee using the criteria in ASME PRA Standard RA-Sa-2009, Section 6. The licensee 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 finds the licensee's conclusion to be reasonable.

3.5.1.8 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 license amendment request. The licensee stated that they are using their on-the-shelf PRA model and did not create a special model to quantify the RHR pump. Accordingly, the licensee used the Rev. 8 Phoenix One-Top Multi-Hazard Model (OTMHM) contained in SNC calculation RIE-PHOENIX-U02. 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 response to APLA-RAI-2, dated April 22, 2021, the licensee clarified that all test and maintenance unavailability probabilities except for those associated with the RHR 2D pump were left at their nominal values for both the increased risk and base cases.

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 peer review standard, and all of the F&Os have been closed. The licensee reviewed the quantification and uncertainty notebooks for each hazard model 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 performedto the PRA model to support the risk analysis for the 2D RHR pump one-time AOT extension.

3.5.1.9 PRA Results and Insights

The licensee provided the table below, to summarize the licensee's calculated incremental conditional core damage probability/incremental conditional large early release probability (ICCDP/ICLERP) for the proposed 15-day AOT for the 2D RHR pump.

Input Parameter	**Value**
CDF_{BASE}	5.78×10^{-5}
CDF_{2D}	8.63×10^{-5}
Delta CDF	2.85×10^{-5}
ICCDP for 15-day LCO	**1.17×10^{-6}**
$LERF_{BASE}$	3.76×10^{-3}
$LERF_{2D}$	3.83×10^{-6}
Delta LERF	7.00×10^{-8}
ICLERP for 15-day LCO	**2.88×10^{-9}**

RG 1.177 provides sets of acceptance criteria for one-time AOT 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 than 1.0×10^{-7} but less than 1.0×10^{-6}, in which case a licensee must implement effective compensatory measures to reduce the sources of increased risk.

For the 2D RHR pump outage, the licensee stated in the LAR that the calculated ICCDP and ICLERP meet the second set of criteria, so it is acceptable, but compensatory measures are therefore required.

3.5.2 <u>Tier 2: Avoidance of Risk-Significant Configurations</u>

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 potentiallyhigh-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 licensee 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 3.7.

3.5.3 <u>Tier 3: Risk-Informed Configuration Risk Management</u>

A Tier 3 program ensures that while an RHR 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 2D RHR pump maintenance activities that would adversely impact risk, and (3) evaluates the impact of maintenance on equipment or systems assumed to remain operable by the RHR pump AOT 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 an RHR 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 RHR pump AOT.

In LAR Attachment 4, dated April 19, 2021, the licensee stated that Hatch has a mature on-line configuration risk management process. The licensee 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.

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

3.5.4 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 AOT 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.

3.6 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 AOTs 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 performedin 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 letter dated April 19, 2021, 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."

Based on the above, the NRC staff concludes that the implementation and monitoring program for the proposed one-time 2D RHR pump AOT extension satisfies the fifth key safety principle of RG 1.177.

3.7 Compensatory Measures

After the expiration of the 7-day CT ends at 9:55 am Eastern Time on April 23, 2021, SNC will establish compensatory measures to help ensure that the remaining ECCS continue to be available to meet all applicable acceptance criteria during a design basis accident or plant transient.

SNC stated that it will implement the following compensatory measures:

- Equipment is protected as required by SNC Procedure NMP-OS-010-002 for 2D RHR pump out-of-service.

- No elective maintenance will be performed on equipment related to the Hatch, Unit 2, ECCS or RCIC System.

- No elective maintenance will be performed on the of U2 emergency diesel generators (EDGs) or the swing (1B) EDG.

In addition, in response to APLA-RAI-1, dated April 22, 2021, the licensee also will implement the following compensatory measures:

- Operations personnel will be briefed on hardened containment vent procedure NMP-OS-019-286 at the beginning of each shift.

- The following components will be protected per NMP-OS-010, and no elective maintenance will be performed.

Component	Description
2P52F1171	2P52F1171 is the isolation valve for the two nitrogen accumulators in the reactor building pneumatic system. This is a normally open manual valve.
2P52F1187	2P52F1187 is the isolation valve from the reactor building pneumatic system to valve 2T48F082. This is a normally open manual valve.
2T48F082	2T48F082 is a normally closed 18-inch air operated valve that is required to open to allow the containment hardened vent path to function
2T48F081	2T48F081 is a normally open 18-inch air operated valve that is required to close to allow the containment hardened vent path to function.
2R25S067	2R25S067 is the Division 2 Critical Instrument Bus. It provides controlpower to the hardened vent system and other loads.

2R25S006	2R25S006 is the 125-volt DC distribution panel providing control power to the 2C diesel generator and to the circuit breaker for the 2B RHR pump and other loads.
2T48D346	2T48D346 is a rupture disc in the hardened vent path line.

- The following controls will be in place for fire area 0040:

  o The room will be protected per NMP-OS-010 (Reference 3).
  o No hot work or transient combustible permits will be issued.

The NRC staff has reviewed the licensee's proposed compensatory measures and concludes, based on information provided by the licensee, that the compensatory measures are appropriate to reduce the risk of unnecessary plant transients, protect systems needed for accident mitigation, and raise operator awareness of necessary recovery actions with one RHR system inoperable.

3.8 TS Changes

For the other RHR system modes such as Suppression Pool Cooling (TS 3.6.2.3), Suppression Pool Spray (TS 3.6.2.4), and Drywell Spray (TS 3.6.2.5), the licensee states that only a single RHR pump in a loop is required to be operable and are thus not affected by this request. The NRC staff reviewed the licensee's applicable TS Bases and UFSAR sections and confirmed that only a single pump in a loop is required for these functions.

The NRC staff reviewed the TS 3.5.1 Condition A proposed actions and CTs. The licensee provided an evaluation which concluded the proposed Condition A actions assure that the plant remains within the bounds of existing design basis safety analyses. The NRC staff reviewed the licensee's evaluation and found it acceptable.

As part of its assessment, the NRC staff considered how these new required actions and CTs relate to the other actions within TS 3.5.1. With regard to how these new required actions and CTs relate to the other actions within TS 3.5.1, the NRC staff notes that these actions and CTs are only applicable to the 2D RHR pump repair until May 1, 2021. The new actions provide the licensee an additional 8 days (15 days from the time of discovery) to repair the RHR pump and are specifically related to the first component of TS 3.5.1, Condition A, "One low pressure ECCS injection/spray subsystem inoperable."

The NRC staff reviewed the editorial changes that moved information to subsequent pages. The NRC staff has determined the editorial changes on pages 3.5-2 through 3.5-4 are acceptable.

3.9 Conclusion

TS LCO 3.5.1 requires the ECCS to be operable. The licensee proposed to be allowed to continue to operate for eight additional days, where the existing TS would have required to go to MODE 3 (Hot Shutdown). The NRC staff concludes, based on information provided by the licensee, that the changes in risk associated with extending the associated CTs are less than that of the guidance thresholds in RG 1.177 and RG 1.174, and, therefore, support the extension of the CTs associated with the inoperability of RHR system from 7 to 15 days. Therefore, the NRC staff concludes that the licensee's proposed new remedial action to be taken when LCO 3.5.1 is not met provides reasonable assurance of public health and safety.

4.0 EMERGENCY CIRCUMSTANCES

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.

On April 16, 2021, at 9:55 a.m., Eastern Time, SNC declared the 2D RHR pump inoperable at Hatch, Unit 2. The 2D RHR pump was started and received an overcurrent alarm. The 2D RHR pump tripped on overcurrent when a restart was attempted.

The required CT for Condition A for TS 3.5.1 is 7 days and the CT will expire on April 23, 2021, at 9:55 a.m. SNC stated that it cannot finish repairs of the 2D RHR by April 23, 2021. SNC has not identified the specific failed component at this time. SNC stated that its engineering fault trees indicate the cause is internal to the pump. SNC estimates that the maintenance tasks associated with the 2D RHR pump will take a total of 15 days. This would be an additional 8 days from the current CT of 7 days.

SNC stated that this situation could not be avoided because the review of previous performance data and preventive maintenance records showed that the 2D RHR pump performed satisfactorily.

SNC declared the 2D RHR pump inoperable on April 16, 2021, and SNC submitted its emergency LAR on April 19, 2021.

Consistent with the guidance of RG 1.177, Revision 1, Element 2: "Perform Engineering Analysis," the NRC staff requested an RAI, dated on April 20, 2021 (ADAMS Accession No. ML21110A684), pertaining to the recent historical operating performance of the RHR redundant train pumps and the residual same train RHR pump. Specifically, those pumps are the 2A, 2B, and 2C, RHR pumps, respectively.

In addition, the NRC staff requested that the licensee provide a review of industry operating experience pertaining to these Ingersoll-Rand pumps.

SNC responded, in part:

> In addition to the 2D pump, a review of the performance data for the other Unit 2 residual heat removal (RHR) pumps (2E11C002A, B, C) was performed. The data were within Hatch and Fleet standards. The review included pump flow, discharge pressure, and differential pressure from dation. Vibration measurements were all in the acceptable range, and there were no significant outliers that would indicate adverse trends with relation to pump performance. …
>
> In addition, industry and pump vendor operating experience (OE) was reviewed,

as well as NRC Information Notices (INs) (e.g., INs 86-39, 87-59, 93-03). This review did not uncover any OE that would suggest a potential common cause mode failure for the 2A, 2B, or 2C RHR pumps based on the 2D pump overcurrent trip.

The NRC staff found the RAI response to be a sufficient evaluation of alternate RHR train reliability and operability to support a reasonable assurance determination.

The NRC staff reviewed the licensee's basis for processing the proposed amendment as an emergency and agrees that an emergency situation exists. The licensee could not have foreseen the complexity of the maintenance tasks, given the considerable pre-planning, training, and pre-staging of equipment and personnel. The NRC staff concludes that the licensee's actions were reasonable and that the identification of the maintenance tasks, the actions of the licensee to inform the NRC staff, and the assistance of a vendor expert with RHR pumps, and the submittal of the emergency LAR, are being addressed in a reasonable amount of time and that the emergency situation could not have been avoided. The NRC staff also concludes that failure to issue this license amendment would result in the shutdown of Hatch, Unit 2.

The NRC staff agrees that an emergency situation exists consistent with the provisions in 10 CFR 50.91(a)(5). 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 involves no significant hazards consideration as discussed below, the NRC staff has determined that a valid need exists for issuance of the license amendment using the emergency provisions of 10 CFR 50.91(a)(5).

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to 10 CFR 50.91(a)(5), if the Commission finds that an emergency situation exists, in that, failure to act in a timely way would result in shutdown of a nuclear power plant, it may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. As noted in Section 4.0 of this SE, the NRC staff has concluded that an emergency situation does exist, in that failure to issue the amendment would result in the shutdown of Hatch, Unit 2. Therefore, a final finding of no significant hazards consideration follows.

The Commission has made a final determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (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.

As required by 10 CFR 50.91(a), in its letter dated April 19, 2021, the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment 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 Technical Specification (TS) 3.5.1 Condition A to allow necessary time to restore the 2D RHR 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 Emergency Core Cooling Systems (ECCS) 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 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 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.

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2 - ISSUANCE OF AMENDMENT NO. 254, REGARDING REVISION TO TECHNICAL SPECIFICATIONS FOR ONE-TIME EXTENSION OF COMPLETION TIMES RELATED TO RESIDUAL HEAT REMOVAL SYSTEM **(EMERGENCY CIRCUMSTANCES)** (EPID L-2021-LLA-0068) DATED APRIL 22, 2021

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Amendment No. ML21109A359

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/DSS/STSB/BC	NRR/DSS/SNSB/BC
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DATE	04/22/2021	04/22/2021	04/22/2021	04/22/2021
OFFICE	NRR/DRA/APLA/BC	OGC - NLO	NRR/LPL2-1/BC	NRR/LPL2-1/PM
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DATE	04/22/2021	04/22/2021	04/22/2021	04/22/2021

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> In addition, during the extended Completion Time the ECCS will remain capable of 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.

Accordingly, the NRC staff has determined that this amendment involves no significant hazards determination.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Georgia official was notified on April 20, 2021, of the proposed issuance of the amendment. The State official had no comments as of the date of issuance of this amendment.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation and 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 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 previously evaluated or, (c) involve a significant reduction in a margin of safety and, therefore, the amendment does not involve a significant hazards consideration; (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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