Southern Nuclear

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant Unit 2
Emergency License Amendment Request for Technical Specification 3.5.1
Regarding One-Time Extension of Completion Time for 2D RHR Pump

Ladies and Gentlemen:

Pursuant to the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Southern Nuclear Operating Company (SNC) hereby requests a proposed license amendment to the Technical Specifications (TS) for Hatch Nuclear Plant (HNP) Unit 2 renewed facility operating license NPF-5. The proposed amendment would revise Condition A of TS 3.5.1, ECCS Operability.

The amendment would allow a one-time increase in the Completion Time from 7 days to 15 days. The increased Completion Time would expire on May 1, 2021 at 0955 eastern daylight time (EDT).

The one-time only change allows for continued repair and testing activities on the 2D Residual Heat Removal (RHR) pump. The expiration date for the proposed allowance is based on the current 7-day Completion Time expiration at 0955 EDT, April 23, 2021, plus the requested additional 8 days (15 days total).

This proposed amendment to the HNP Unit 2 TS is being requested on an emergency basis for the Unit 2 RHR System, pursuant to 10 CFR 50.91(a)(5). The Unit 1 RHR System is not affected by this proposed amendment.

SNC requests approval of the proposed license amendment as soon as possible and no later than April 22, 2021 based on emergent circumstances at HNP Unit 2 in accordance with the provisions of 10 CFR 50.91(a)(5). A discussion of the emergency situation is provided in the enclosure to this letter. The amendment, if approved, will be implemented immediately upon issuance.

The enclosure provides a description and assessment of the proposed change, including a no significant hazards considerations analysis, regulatory requirements, and environmental considerations. Attachments 1 and 2 contain marked-up TS pages and revised TS pages, respectively, reflecting the proposed changes. Attachment 3 contains a markup of the TS

Bases, for information only. Attachment 4 contains an evaluation of the risk impact and a discussion of the compensatory measures related to the changes in this amendment request.

In accordance with the SNC administrative procedures and the HNP quality assurance program manual, this proposed license amendment has been reviewed and approved by the plant review board.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia of this license amendment request by transmitting a copy of this letter, enclosure, and attachments to the designated State Official.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 19th day of April 2021.

Respectfully submitted,

Cheryl A. Gayheart
Regulatory Affairs Director

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Enclosure: Description and Assessment of the Proposed Change

Attachments:

1. HNP Unit 2 Technical Specification Marked-up Page
2. HNP Unit 2 Revised Technical Specification Page
3. HNP Unit 2 Technical Specification Bases Marked-up Pages (information only)
4. Evaluation of Risk Impact and Compensatory Measures

cc: NRC Regional Administrator, Region II
NRC NRR Project Manager – Hatch
NRC Senior Resident Inspector – Hatch
Director, Environmental Protection Division - State of Georgia
RType: CHA02.004

Edwin I. Hatch Nuclear Plant Unit 2

Emergency License Amendment Request for Technical Specification 3.5.1 Regarding One-Time Extension of Completion Time for 2D RHR Pump

Enclosure

Description and Assessment of the Proposed Change

1. Summary Description

The proposed amendment to Hatch Nuclear Plant (HNP) Unit 2 renewed facility operating license NPF-5 would revise Technical Specification (TS) 3.5.1, “ECCS Operability,” Condition A, “One low pressure ECCS injection/spray subsystem inoperable; OR One LPCI pump in both LPCI subsystems inoperable.” This request is specifically related to the first component of Condition A, “One low pressure ECCS injection/spray subsystem inoperable.”

The change would allow a one-time increase in the Completion Time from 7 days to 15 days. The allowance of the extra 8 days would only be applicable while the compensatory measures described in Section 3.3 and as affirmed in the NRC’s safety evaluation are implemented, and would expire on May 1, 2021, at 0955 EDT.

On April 16, 2021, at 0955 EDT, HNP Unit 2 RHR pump D was declared inoperable due to an overcurrent alarm with a subsequent failed restart (trip on overcurrent). Southern Nuclear Operating Company (SNC) has been performing testing and repair activities and the pump vendor is on site assisting with the investigation.

The comprehensive repair work has been time-consuming; however, SNC has demonstrated due diligence by safely performing testing and maintenance activities around the clock. SNC now expects that repair and testing will extend past the 7-day Completion Time of the above-listed TS and therefore requests additional time to make careful, prudent repairs with appropriate compensatory measures in place to return the HNP Unit 2 RHR pump 2D to Operable status. To provide allowance for additional time to complete repairs, SNC is requesting the Completion Time be temporarily extended from 7 days to 15 days.

Note that while the RHR System functions in several modes of operation, as discussed in Section 2.2, 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. Detailed Description

2.1 Emergency Circumstances

Why the Condition Occurred:

The emergency circumstances resulted from the unforeseen failure of the HNP Unit 2 RHR pump 2D during unit operation. On April 16, 2021, at 0955 EDT, HNP Unit 2 RHR pump 2D was declared inoperable due to an overcurrent alarm with a subsequent failed restart (trip on overcurrent). SNC has been performing testing and repair activities and the pump vendor is on site assisting with the investigation.

The required Completion Time for Condition A for TS 3.5.1 of seven days is currently applicable and will expire on April 23, 2021 at 0955 EDT. SNC cannot finish repairs of

RHR pump 2D by that time. Neither a routine nor an exigent amendment can be processed prior to April 23, 2021 at 0955 EDT.

At the time the repair work commenced, the cause of failure of the 2D RHR pump was unknown. Subsequent troubleshooting efforts indicated that pump dismantlement was required to identify the exact cause of the failure. While SNC has not identified the specific failed component at this time, the engineering fault trees indicate the cause is internal to the pump.

Why this Situation Could Not be Avoided:

A review of the performance data for the 2E11C002D (i.e., “2D”) RHR pump was performed and all reviewed data were within Hatch and Fleet standards. A review of the pump flow, discharge pressure, and differential pressure from 2016 to present showed a steady flat trend with no indication of a degradation. Reviewing the pump vibration from 2019 to present showed all values within the acceptable range.

During the annual pump/motor review preventive maintenance (PM) that was performed on 4/20/2020, all of the parameters indicative of pump health (i.e. flow, differential pressure and vibration) were within acceptable values. In addition, visual inspection of the pump area and surroundings reveal no signs of degradation of the pump’s mechanical seals. On January 21, 2021 the quarterly 2D pump in-service test (IST) was performed. This test and the previous IST were both performed satisfactorily. During the January IST run, vibration measurements were taken and all the levels met SNC Fleet vibration standards.

The data for the 2D pump motor were also reviewed as part of the annual pump/motor review PM. All environmental qualification (EQ) PM (i.e. thermography, EMI, oil change/ sampling, megger, EMAX) has been performed on the 2D RHR motor by required dates. No degraded conditions or parameters outside acceptable values were observed. In addition, on 1/20/2021, the motor oil samples were analyzed and the results were satisfactory.

Summary of Emergency Circumstances:

In summary, the emergency circumstances resulted from the unforeseen failure of the HNP Unit 2 RHR pump 2D during unit operation. The required Completion Time for Condition A for TS 3.5.1 of seven days is currently applicable and will expire on April 23, 2021 at 0955 EDT. SNC cannot finish repairs of the RHR pump 2D by that time. Neither a routine nor an exigent amendment can be processed prior to April 23, 2021 at 0955 EDT.

SNC has performed due diligence by safely performing testing and maintenance activities around the clock. The repair plan included many provisions to ensure timely execution of the work including the use of experienced personnel, pre-assembled components, and pre-staging of equipment. An experienced pump vendor is on site. Therefore, efforts were made to minimize the likelihood for delays due to job planning or preparation. Contingencies were developed and carried out for the existing problems.

SNC requests an expedited review of the proposed license amendment in accordance with the provisions of 10 CFR 50.91(a)(5) based on avoiding the need to shut down HNP Unit 2 without an approved amendment. If the proposed license amendment is not approved, Unit 2 will be required to enter shutdown TS 3.5.1 Condition B on April 23, 2021 at 0955 EDT.

On the basis of the discussion herein, SNC has determined that emergency circumstances exist, has used its best efforts to make a timely application, and did not knowingly cause the emergent situation.

2.2 System Design and Operation

Residual Heat Removal System

The RHR system is a system of pumps, heat exchangers, and piping that fulfills the following functions:

- Removal of decay and sensible heat from the reactor core during and after shutdown.
- Removal of stored and decay heat from the reactor core following a design basis loss of coolant accident (LOCA).
- Removal of heat from the primary containment following a LOCA to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the water inside the primary containment. The redundancy of the equipment provided for containment cooling is further extended by a separate part of the RHR system which sprays cooling water into the containment.

From a safety design basis perspective, the RHR system is designed:

- In the low pressure coolant injection (LPCI) mode to act automatically, in combination with other ECCS systems, to restore and maintain the coolant inventory in the RPV so that the core is adequately cooled to preclude fuel-cladding perforation and subsequent energy release due to a metal-water reaction.
- In the containment spray mode to remove airborne particulates in the drywell and to reduce the temperature and pressure of the primary containment atmosphere post-LOCA.
- So that a source of water for restoration of reactor vessel coolant inventory is located within the primary containment in such a manner that a closed cooling water path is established.
- To provide a high degree of assurance that the RHR system operates satisfactorily during a LOCA and that each active component is capable of being tested during operation of the nuclear system.
- To satisfy Seismic Category I requirements.
- To satisfy applicable environmental qualification requirements.

- So that the residual heat removal service water (RHRSW) can be pumped directly into the RHR system.
- To provide heat exchangers with a heat removal capability for long-term containment cooling.

From a power generation design basis perspective, the RHR system is designed:

- To have enough heat removal capacity to cool down the reactor to 125ºF within 20 hours after shutdown.
- To have fuel pool connections so that the RHR heat exchangers can be used to supplement the fuel pool cooling capacity.
- So that a closed loop flow path between the suppression pool and the RHR heat exchangers can be established to utilize the heat removal capability of these heat exchangers to cool the suppression pool.

The RHR system is designed for six modes of operation to satisfy all the objectives and bases. The modes are summarized as follows:

Mode	Action	Function
LPCI (ECCS)	Accident safety	Restore and maintain reactor vessel water level after a LOCA.
Containment spray	Post-accident safety	Remove airborne particulates in the drywell and limit temperature and pressure in the torus and drywell after a LOCA.
Suppression pool cooling	Abnormal operation	Remove heat from the suppression pool water.
Shutdown cooling	Planned operation	Remove decay and residual heat from the reactor core to achieve and maintain a cold shutdown condition.
Minimum flow	Equipment protection	Prevent pump damage when operating against closed discharge valve.
Test	System test	Test RHR system during plant operation.

Note that while the RHR System functions in several modes of operation, SNC has determined that only the 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.

The major equipment of the RHR system consists of two heat exchangers and four RHR pumps. The RHR Service Water system provides cooling water to the heat exchangers.

The equipment is connected by associated valves and piping, while controls and instrumentation are provided for proper system operation.

The RHR pumps are sized for the flow required during LPCI operation, which is the subsystem that requires the maximum flowrate. The pumps are arranged and located so that adequate suction head is ensured for all operating conditions. Each pump motor is air cooled.

The heat exchangers are sized on the basis of their required duty for the shutdown cooling function. The heat exchanger shell and tube sides are provided with drain connections. The shell side is provided with a vent to remove noncondensable gases. Relief valves on the heat exchanger shell inlets and a relief valve on the HPCI steam supply line to the RHR heat exchangers protect the heat exchangers from overpressure.

With the unit in Mode 5, the RHR system can be connected to the fuel pool cooling and cleanup system (FPCCS), so that the RHR heat exchangers can assist fuel pool cooling during high heat-load conditions.

One loop, consisting of a heat exchanger, two RHR pumps in parallel, and associated piping, is located in one area of the reactor building. The remaining heat exchanger, pumps, and piping, all of which form a second loop, are located in another area of the reactor building to minimize the possibility of a single physical event causing the loss of the entire system.

Emergency Core Cooling System

The ECCS is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss of coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network consists of the High Pressure Coolant Injection (HPCI) System, the Core Spray (CS) System, the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System, and the Automatic Depressurization System (ADS). The suppression pool provides the required source of water for the ECCS. Although no credit is taken in the safety analyses for the condensate storage tank (CST), it is capable of providing a source of water for the HPCI and CS Systems.

On receipt of an initiation signal, ECCS pumps automatically start. Simultaneously, the system aligns and the pumps inject water, taken either from the CST or suppression pool, into the Reactor Coolant System (RCS) as RCS pressure is overcome by the discharge pressure of the ECCS pumps. Although the system is initiated, ADS action is delayed, allowing the operator to interrupt the timed sequence if the system is not needed. The HPCI pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the vessel to cool the core. If the break is small, the HPCI System will maintain coolant inventory as well as vessel level while the RCS is still pressurized. If HPCI fails, it is backed up by ADS in combination with LPCI and CS. In this event, the ADS timed sequence could be allowed to time out and open the selected safety/relief valves (S/RVs) depressurizing the RCS, thus allowing LPCI and CS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly and the LPCI and CS cool the core.

Water from the break returns to the suppression pool where it is re-used. Water in the suppression pool may be circulated through a heat exchanger cooled by the RHR Service Water System. Depending on the location and size of the break, portions of the ECCS may be ineffective; however, the overall design is effective in cooling the core regardless of the size or location of the piping break.

All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS equipment.

The CS System is composed of two independent subsystems. Each subsystem consists of a motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suppression pool to the sparger. The CS System is designed to provide cooling to the reactor core when reactor pressure is low. Upon receipt of an initiation signal, the CS pumps in both subsystems are automatically started when AC power is available. When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems, each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop. The two LPCI subsystems can be interconnected via the RHR System cross-tie valve; however, the cross-tie valve is maintained closed with its power removed to prevent loss of both LPCI subsystems during a LOCA. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an automatic initiation signal, all LPCI pumps will start immediately if power is provided by the primary feeds from Startup Auxiliary Transformer (SAT) 2C, 2D, and 2E. If power for either Bus 2E or Bus 2F is provided by the alternate feeds from the SATs or by the DGs, the engineered safety feature (ESF) Division I motors have a time delay. If power for either Bus 2F or Bus 2G is provided by the alternate feeds from the SATs or by the DGs, the engineered safety feature (ESF) Division II motors have a time delay. If a time delay is active for the associated motor, LPCI pump C starts within 1 second when power is available and the LPCI A, B, and D pumps are started after a 10 second delay.

RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the recirculation loops. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the corresponding recirculation loop, begins. The water then enters the reactor through the jet pumps. Full flow test lines are provided for each division (two test lines total) to route water from the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (162 psid to 1200 psid, vessel to pump suction). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using a "keep fill" system (jockey pump system). The HPCI System is normally aligned to the CST.

The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water. Therefore, HPCI does not require a "keep fill" system.

The ADS consists of 7 of the 11 S/RVs. It is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (CS and LPCI), so that these subsystems can provide coolant inventory makeup. Each of the S/RVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves.

2.3 Current Technical Specification Requirements

Currently, Condition A of TS 3.5.1 requires restoration of an inoperable RHR pump to operable status within 7 days.

2.4 Reason for the Proposed Change

Despite diligent and prudent efforts, SNC has been unable to return the HNP Unit 2 RHR pump 2D to operable service and now expects it will be unable to do so by expiration of the Completion Time for TS 3.5.1, Condition A (i.e., 0955 EDT, April 23, 2021). The table below summarizes the maintenance tasks to be performed for the 2D RHR pump and the duration of each task. Please note that the durations below include contingencies; thus, SNC has confidence that the 2D RHR pump will be restored within this timeframe. In addition, SNC will work to restore the 2D RHR pump to Operable status as soon as reasonably and safely achievable, regardless of the additional Completion Time duration.

Task No.	Project Task	Duration (hours)
1*	Inspect Pump; Determine if Pump and Motor Lift Required	74
2	Remove Interferences	27
3	Prepare for Pump Disassembly	28
4	Disassemble Pump, Inspect, and Repair	112
5	Reassemble Pump	55
6	Remove Tagouts; Restore Pump to Operable	44
TOTAL:		340

*Task already performed.

To allow for additional unforeseen discoveries, an additional 20 hours is requested (360 hours total).

Approval of the proposed change would allow SNC to continue repair, refurbishment, and replacement activities as necessary and without undue risk as demonstrated in Attachment 4 of this license amendment request.

2.5 <u>Description of the Proposed Change</u>

The following changes are proposed to the HNP Unit 2 TS:

- The following revision is proposed to TS 3.5.1 (added text in *blue italics*):

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.	7 days
	<u>OR</u>	
	----------------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-21-0411, dated April 19, 2021.	*7 days*
	<u>AND</u>	
	A.2.2 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	*15 days*

The one-time only change allows for continued repair and testing activities in a prudent fashion. It also allows for appropriate compensatory measure to established prior to the expiration of the current 7-day Completion Time prior to entering the extended Completion Time. The expiration date of May 1, 2021, 0955 EDT is based on the current 7-day Completion Time expiration at 0955 EDT, April 23, 2021, plus the requested additional 8 days (15 days total). The allowance would only apply as long as the compensatory measures described in Section 3.3 of this application are implemented.

3. Technical Evaluation

3.1 <u>Defense-in-Depth</u>

Should an event occur requiring the ECCS, the remaining low pressure ECCS pumps are capable of performing the safety function. In addition, TS 3.5.1 Condition A is applicable when two RHR pumps are inoperable, whether two inoperable RHR pumps are in the same LPCI subsystem or whether there is one inoperable RHR pump in each LPCI subsystem, whereas this request is only applicable to one RHR pump being inoperable.

In addition, compensatory measures, as described in Section 3.3, will be in place and available.

3.2 Safety Margin Evaluation

The proposed TS change is consistent with the principle that sufficient safety margins are maintained based on the following:

Codes and standards (e.g., American Society of Mechanical Engineers (ASME), Institute of Electrical and Electronic Engineers (IEEE) or alternatives approved for use by the NRC) are met. The proposed change is not in conflict with approved codes and standards relevant to the RHR System.

The ECCS system has sufficient capacity to function for design basis events while in Condition A. Assuming no additional failures, the 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. In addition, Condition A is applicable when two RHR pumps are inoperable, whether two inoperable RHR pumps are in the same LCPI subsystem or whether there is one inoperable RHR pump in each LPCI subsystem, whereas this request is only applicable to one RHR pump being inoperable.

3.3 Compensatory Measures

The following compensatory measures will help ensure that ECCS remains available to meet all applicable acceptance criteria during a design basis accident or plant transient:

- Equipment is protected as required by SNC Procedure NMP-OS-010-002 (Reference 1) for 2D RHR pump out-of-service.
- No elective maintenance will be performed on equipment related to the U2 ECCS or RCIC System.
- No elective maintenance will be performed on the of U2 emergency diesel generators (EDGs) or the swing (1B) EDG.

3.4 Maintenance Rule Control

The RHR pumps are included under the HNP Maintenance Rule Program and function as described in Section 2.2. The RHR pumps are monitored for unavailability as part of the Maintenance Rule performance monitoring. As part of compliance with 10 CFR 50.65, performance is monitored against licensee-established goals. If the performance of the RHR System does not meet the established goals, 10 CFR 50.65(a)(1) requires appropriate corrective action to be taken to restore the system's performance to an acceptable level.

The pumps' reliability is tracked by quarterly in-service testing (IST). If, during testing, pump parameters are outside of the established criteria of the IST program, the IST program requires action to address the situation.

3.5 Evaluation of Risk Impacts

The risks associated with a one-time extension of the HNP Unit 2 TS 3.5.1, “ECCS - Operating,” Condition A, to allow a one-time increase in the Completion Time from 7 days to 15 days have been evaluated by way of probabilistic risk assessment (PRA) models that meet all scope and quality requirements in NRC Regulatory Guide (RG) 1.200, Revision 2 (Reference 2).

This plant-specific risk assessment followed the guidance in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3 (Reference 3), and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," Revision 1 (Reference 4).

Attachment 4 of this license amendment request presents the evaluation of risk impacts due to the proposed amendment.

4. Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

The following regulatory requirements have been considered:

10 CFR 50.36:

10 CFR, Section 50.36, "Technical specifications," in which the Commission established its regulatory requirements related to the contents of the TS. Specifically, 10 CFR 50.36(c)(2) states, in part, "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility." The proposed changes do not affect compliance with these regulations.

10 CFR 50.46:

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" in which the Commission established its regulatory requirements related to core cooling: The proposed changes maintain the acceptance criteria of this section.

10 CFR 50 Appendix A:

The applicable 10 CFR Part 50, Appendix A, "General Design Criteria 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.

4.2 Precedent

The proposed change is similar to NRC-approved License Amendment 230 issued under emergency circumstances to Columbia Generating Station on February 1, 2015 (NRC Agencywide Documents Access and Management System (ADAMS) Accession No. ML15030A501). This amendment makes a one-time revision to TS 3.5.1, "ECCS [Emergency Core Cooling System]- Operating," TS 3.6.1.5, "Residual Heat Removal (RHR) Drywell Spray," and TS 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," to extend the Completion Time (CT) of Required Actions specifically associated with RHR System B inoperability from 7 days to 14 days. The accompanying NRC safety evaluation concluded the license amendment involved no significant hazards consideration.

4.3 No Significant Hazards Consideration Analysis

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests an amendment to Hatch Nuclear Plant (HNP) Unit 2 renewed facility operating license NPF-5. The proposed amendment would revise TS 3.5.1 Condition A to allow additional time to restore the 2D residual heat removal (RHR) pump to OPERABLE status and would expire on May 1, 2021.

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed 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 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.

   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.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed herein, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. Environmental Consideration

SNC has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types, or significant increase in the amounts, of any effluent that may be released off site, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. References

1. SNC Procedure NMP-OS-010-002, *Hatch Protected Equipment Logs*, Version 11.1.
2. Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2, dated March 2009.
3. Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, Revision 2, dated April 2015.
4. Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications*, Revision 1, dated May 2011.

Edwin I. Hatch Nuclear Plant Unit 2

Emergency License Amendment Request for Technical Specification 3.5.1 Regarding One-Time Extension of Completion Time for 2D RHR Pump

Attachment 1

HNP Unit 2 Technical Specification Marked-up Page

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-21-0411, dated April 19, 2021. *AND*	*7 days*
	A.2.2 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	*15 days*

(continued)

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Emergency License Amendment Request for Technical Specification 3.5.1
Regarding One-Time Extension of Completion Time for 2D RHR Pump

Attachment 2

HNP Unit 2 Revised Technical Specification Page

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. <u>OR</u> One LPCI pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days
	<u>OR</u>	
	----------------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-21-0411, dated April 19, 2021. <u>AND</u>	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.	1 hour
	<u>AND</u>		
	C.2	Restore HPCI System to OPERABLE status.	14 days
D. HPCI System inoperable. <u>AND</u> Condition A entered.	D.1	Restore HPCI System to OPERABLE status.	72 hours
	<u>OR</u>		
	D.2	Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	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.	12 hours
	<u>AND</u>		
	F.2	Reduce reactor steam dome pressure to ≤ 150 psig.	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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Emergency License Amendment Request for Technical Specification 3.5.1
Regarding One-Time Extension of Completion Time for 2D RHR Pump

Attachment 3

HNP Unit 2 Technical Specification Bases Marked-up Pages (information only)

ACTIONS

A.1 (continued)

based on a reliability study (Ref. 12) that evaluated the impact on ECCS availability, assuming various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

A.2.1 and A.2.2

The Completion Time to restore the low pressure ECCS injection/spray subsystem to OPERABLE status to facilitate the 2D RHR pump repair may be extended to 15 days, provided action is taken within 7 days to establish compensatory and risk management controls.

The A.2.1 and A.2.2 Required Actions are modified by two Notes. Note 1 ensures that the A.2.1 and A.2.2 Required Actions are only applied during the 2D RHR pump repair. Note 2 limits the time period the A.2.1 and A.2.2 Required Actions may be used.

The extended Completion Time is subject to additional compensatory controls specified in Enclosure 1 Section 3.3 of SNC letter NL-21-0411, dated April 19, 2021, that consist of controls that must be established and maintained during the extended Completion Time period to preserve defense-in-depth.

If Required Action A.2.1 is met, the allowed time to restore the ECCS injection/spray subsystem to OPERABLE status can be extended to 15 days from entry into Condition A. The extended Completion Time of Required Action A.2.2 represents a balance between the risk associated with continued plant operation with less than the required system or component redundancy and the risk associated with initiating a plant transient while transitioning the unit based on the loss of redundancy. With compensatory and risk management controls established, the remaining OPERABLE ECCS injection/spray subsystems are adequate to provide low pressure coolant injection to the reactor core. The extended Completion Time takes into account the low probability of a DBA or an LOSP occurring during this period.

The Completion Time of Required Action A.2.2 is based on a defense-in-depth philosophy, and is risk informed using the plant PRA. The risk impact of the extended Completion Time has been evaluated pursuant to the risk assessment and management provisions of the Maintenance Rule, 10 CFR 50.65(a)(4), and the associated implementation guidance, Regulatory Guide 1.160. Regulatory Guide 1.160 endorses the guidance in Section 11 of NUMARC 93-01. This guidance provides for the consideration of dynamic plant configuration

(continued)

issues, emergent conditions, and other aspects pertinent to plant operation with the 2D RHR pump inoperable for an extended period of time. These considerations may result in additional risk management and other compensatory actions being required during the extended period that the 2D RHR pump is inoperable.

B.1

If the inoperable low pressure ECCS subsystem cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.

Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4, because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.

Required Action B.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If the HPCI System is inoperable and the RCIC System is verified to be OPERABLE, the HPCI System must be restored to OPERABLE

(continued)

Edwin I. Hatch Nuclear Plant Unit 2

Emergency License Amendment Request for Technical Specification 3.5.1 Regarding One-Time Extension of Completion Time for 2D RHR Pump

Attachment 4

Evaluation of Risk Impact and Compensatory Measures

1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to assess the acceptability, from a risk perspective, of a change to extend the Hatch completion time (CT) for Tech Spec Condition 3.5.1.A from 7 days to 15 days for Unit 2 in order to allow for repair of the 2D RHR Pump. These proposed changes are requested to be effective only during a one-time extension.

1.2 BACKGROUND

1.2.1 Technical Specification Changes

Since the mid-1980s, the NRC has been reviewing and granting improvements to TS that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993, the NRC stated that it . . .

> . . . expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. . . Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.

The NRC reiterated this point when it issued the revision to 10 CFR 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decision-making and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements.
3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.
4. The Commission's safety goals and subsidiary numerical objectives are to be used with consideration of uncertainties in making regulatory judgments…

The movement of the NRC to more risk-informed regulation has led to the NRC identifying Regulatory Guides and associated processes by which licensees can submit changes to the plant design basis including Technical Specifications. These guides are discussed in the following section.

1.3 REGULATORY GUIDES

Three Regulatory Guides provide primary inputs to the evaluation of a Technical Specification change. Their relevance is discussed in this section.

1.3.1 Regulatory Guide 1.200, Revision 3

Regulatory Guide 1.200, Revision 3 [Ref. 1] describes an acceptable approach for determining whether the quality 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 light-water reactors. This guidance is intended to be consistent with the NRC's PRA Policy Statement and more detailed guidance in Regulatory Guide 1.174.

It is noted that RG 1.200, Revision 3 endorses Addendum A of the ASME/ANS PRA Standard [Ref. 4] as clarified in Appendix A of RG 1.200, Revision 3.

1.3.2 Regulatory Guide 1.174, Revision 3

Regulatory Guide 1.174 [Ref. 2] specifies an approach and acceptance guidelines for use of PRA in risk informed activities. RG 1.174 outlines PRA related acceptance guidelines for use of PRA metrics of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) for the evaluation of <u>permanent</u> TS changes. The guidelines given in RG 1.174 for determining what constitutes an acceptable permanent change specify that the ΔCDF and the ΔLERF associated with the change should be less than specified values, which are dependent on the baseline CDF and LERF, respectively.

RG 1.174 also specifies guidelines for consideration of external events. External events can be evaluated in either a qualitative or quantitative manner.

Since this LAR is for a one-time TS change, the ΔCDF and the ΔLERF of RG 1.174 do not specifically apply.

1.3.3 Regulatory Guide 1.177 Revision 2

Regulatory Guide 1.177 [Ref. 3] specifies a risk-informed approach and acceptance guidelines for the evaluation of plant technical specification changes. RG 1.177 identifies a three-tiered approach for the evaluation of the risk associated with a proposed TS change as identified below:

- Tier 1 is an evaluation of the plant-specific risk associated with the proposed TS change, as shown by the change in core damage frequency (CDF) and incremental conditional core damage probability (ICCDP). Where applicable, containment performance should be evaluated on the basis of an analysis of large early release frequency (LERF) and incremental conditional large early release probability (ICLERP). The acceptance guidelines given in RG 1.177 for determining an acceptable <u>permanent</u> TS change are that the

ICCDP and the ICLERP associated with the change should be less than 1E-06 and 1E-07, respectively. RG 1.177 also addresses risk metric requirements for one-time TS changes, as outlined in Section 1.3.4 (Acceptance Guidelines) of this risk assessment.

- Tier 2 identifies and evaluates, with respect to defense-in-depth, any potential risk-significant plant equipment outage configurations associated with the proposed change. The licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed TS change is out-of-service.
- Tier 3 provides for the establishment of an overall configuration risk management program (CRMP) and confirmation that its insights are incorporated into the decision-making process before taking equipment out-of-service prior to or during the CT. Compared with Tier 2, Tier 3 provides additional coverage based on any additional risk significant configurations that may be encountered during maintenance scheduling over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance, testing, and corrective and preventive maintenance.

This risk analysis supports the three tiers of RG 1.177, specifically the comparison of the results with the acceptance guidelines for ICCDP and ICLERP associated with changing a Technical Specification Completion Time, the assessment of risk-significant combinations, and the use of the Configuration Risk Management Program.

1.3.4 Acceptance Guidelines

Risk significance in a LAR is determined by comparison of changes in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) and values of Incremental Conditional Core Damage Probability (ICCDP) and Incremental Conditional Large Early Release Probability (ICLERP) produced by a permanent change to either the plant design basis or Technical Specifications to the guidelines given in Regulatory Guide 1.174 and Regulatory Guide 1.177. Reg. Guide 1.174 specifies the acceptable changes in CDF and LERF for permanent changes. Reg. Guide 1.177 specifies the acceptable ICCDP and ICLERP for permanent changes, usually associated with changing CT.

Reg. Guide 1.177 directly addresses the risk metric requirements for one-time TS changes, as reproduced below:

> *"For one-time only changes to TS CTs, the frequency of entry into the CT may be known, and the configuration of the plant SSCs may be established. Further, there is no permanent change to the plant CDF or LERF, and hence the risk guidelines of Regulatory Guide 1.174 cannot be applied directly. The following TS acceptance guidelines specific to one-time only CT changes are provided for evaluating the risk associated with the revised CT:*

1. *The licensee has demonstrated that implementation of the one-time only*

   TS CT change impact on plant risk from implementing the one time-only TS CT change is acceptable (Tier 1):

   - *An ICCDP of less than $1.0x10^{-6}$ and an ICLERP of less than $1.0x10^{-7}$, or*
   - *An ICCDP of less than $1.0x10^{-5}$ and an ICLERP of less than $1.0x10^{-6}$ with effective compensatory measures implemented to reduce the sources of increased risk.*

2. *The licensee has demonstrated that there are appropriate restrictions on dominant risk-significant configurations associated with the change (Tier 2).*
3. *The licensee has implemented a risk-informed plant configuration control program. The licensee has implemented procedures to utilize, maintain, and control such a program (Tier 3)."*

Based on the available quantitative guidelines for other risk-informed applications, it is judged that the quantitative criteria shown in Table 1-1 represent a reasonable set of acceptance guidelines. For the purposes of this evaluation, these guidelines demonstrate that the risk impacts are acceptably low. This, combined with effective compensatory measures to maintain lower risk, will ensure that the TS change meets the intent of small risk increases consistent with the Commission's Safety Goal Policy Statement.

Table 1-1

PROPOSED RISK ACCEPTANCE GUIDELINES

RISK ACCEPTANCE GUIDELINE	BASIS
ICCDP < 1E-6, or ICCDP < 1E-5 with effective compensatory measures implemented to reduce the sources of increased risk	ICCDP is an appropriate metric for assessing risk impacts of out of service equipment per RG 1.177. This guideline is specified in Section 2.4 of RG 1.177.
ICLERP < 1E-7, or ICLERP < 1E-6 with effective compensatory measures implemented to reduce the sources of increased risk	ICLERP is an appropriate metric for assessing risk impacts of out of service equipment per RG 1.177. This guideline is specified in Section 2.4 of RG 1.177.

1.4 SCOPE

This section addresses the requirements of RG 1.200, Revision 3 Section 3.1 which directs the licensee to define the treatment of the scope of risk contributors (i.e., internal initiating events, external initiating events, and modes of power operation at the time of the initiator). Discussion of these risk contributors are as follows:

- Internal Events (IE) – The Hatch PRA model used for this analysis includes a full range of internal initiating events for at-power configurations. The IE model is further discussed in Section 1.5.
- Internal Flooding (IF) – The Hatch PRA model used for this analysis includes flooding scenarios. The IF model is further discussed in Section 1.5.
- Low Power Operation – The intent is for the unit to remain at power during the completion time. Since the RHR is used for shutdown cooling, there is some risk involved with going into lower modes; however, that is not quantified or discussed any further in this assessment.
- Shutdown / Refueling – Hatch does not have a shutdown PRA model, but instead relies upon deterministic methodology to assess defense-in-depth of key safety functions. The intent is for the unit to remain at-power for the duration of the extended CT. Hatch TS 3.4.7, 3.4.8 and 3.9.7 have separate requirements associated with shutdown cooling when the unit is not online. Note that RHR is the key heat removal method in shutdown.
- Internal Fires – The Hatch PRA model used for this analysis contains an as-built, as-operated Fire PRA model. The Fire PRA model is discussed further in Section 1.5.
- Seismic - Hatch has a Seismic PRA. However, the results are not sensitive to maintenance on the RHR pump and therefore the quantified results are not used. The sensitivity is further discussed in Section 1.5.
- Other External Events - Other external event risks (including external flooding and high winds) were assessed in the Hatch Other External Events Screening calculation [Ref. 10] and screened from the PRA.

1.5 Hatch PRA MODELS

This section addresses the requirements of Section 3.1 of RG 1.200, Revision 3 [Ref. 1] which directs the licensee to identify the portions of the PRA used in the analysis.

The PRA analysis uses the Rev. 8 Phoenix One-Top Multi-Hazard Model contained in SNC calculation RIE-PHOENIX-U02 [Ref. 5]. 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. 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 delta risk calculations for a one-time TS change should not be impacted. If a permanent change were being requested, model issues could impact the overall CDF and LERF and would need to be addressed further.

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. A review of the quantification and uncertainty notebooks for each hazard model did not find any assumption or uncertainty that would impact the results of this evaluation.

2.0 RISK ANALYSIS

This section evaluates the plant-specific risk associated with the proposed TS change, based on the risk metrics of CDF, ICCDP, LERF, and ICLERP.

2.1 ASSESSMENT OVERVIEW AND ASSUMPTIONS

2.1.1 Overview

This analysis is performed for unavailability of 2D RHR Pump. The PRA analysis involves identifying the system and components or maintenance activities modeled in the PRA which are most appropriate for use in representing the extended CT configurations and comparing the results to the baseline. Table 2.1-1 lists the base risk metrics for the Full Power Internal Events (FPIE) PRA, internal flooding PRA, and the Fire PRA (FPRA).

Table 2.1-1

HATCH CDF AND LERF BASE RISK METRICS

Hazard(s)	Risk (1/yr)
OTMHM CDF	6.10E-05
OTMHM LERF	3.76E-06

The general configuration for the extended CT is Hatch at-power on both units with the 2D RHR Pump out of service. The risk impact is for Unit 2. The planned maintenance is expected to focus on repairing the pump within the requested extended CT. Concurrent maintenance work will be carefully managed during the extended CT, through the use of the Configuration Risk Management Program.

The PRA model was quantified using the base “average test and maintenance” PRA model with the 2D RHR Pump maintenance and random failure events set to 1.0. The average test and maintenance model represent baseline assumed maintenance frequencies for all components except for Technical Specification violations that are normally excluded in the disallowed maintenance (mutually exclusive) logic in the base PRA model. As a conservative measure, maintenance events for equipment that is protected per site processes during the RHR 2D outage were left at their normal values. Adjustments for common cause factors associated with the RHR 2D pump are also included.

Table 2.1-2

EXTENDED CT CONFIGURATION REPRESENTATION

BASIC EVENT / GATE	DESCRIPTION	VALUE
CC-RS-4_U2	1/4, P4SS2E11C002D	1.0
CC-RS-49_U2	1/4, P4SR2E11C002D	1.0
MNUNRS_TRNB_U2	2D RHR Pump Maintenance	T
CC-RS-7_U2	2/4, P4SS22E11C0022C P4SS22E11C0022D	8.92E-03
CC-RS-9_U2	2/4, P4SS2E11C0022D P4SS1E11C002A	8.92E-03
CC-RS-10_U2	2/4, P4SS1E11C002D P4SS1E11C002B	8.92E-03
CC-RS-12_U2	3/4, P4SS2E11C002C P4SS2E11C002D P4SS2E11C002A	1.05E-03
CC-RS_13_U2	3/4, P4SS2E11C002B P4SS2E11C002D P4SS2E11C002C	1.05E-03
CC-RS-14_U2	3/4, P4SS2E11C002D P4SS2E11C002A P4SS2E11C002B	1.05E-03
CC-RS-15_U2	4/4, P4SS2E11C002B P4SS2E11C002D P4SS2E11C002A P4SS2E11C002C	1.67E-03
CC-RS-52_U2	2/4, P4SR2E11C002C P4SR2E11C002D	5.10E-03
CC- RS-54_U2	2/4, P4SR2E11C002A P8R2E11C002D	5.10E-03
CC- RS-55_U2	2/4, P4SR2E11C002D P4SR2E11C002B	5.10E-03
CC- RS-58_U2	3/4, P4SR2E11C002C P4SR2E11C002D P4SR2E11C002B	9.26E-04
CC- RS-57_U2	3/4, P4SR2E11C002C P4SR2E11C002A P4SR2E11C002D	9.26E-04
CC- RS-59_U2	3/4, P4SR2E11C002A P4SR2E11C002D P4SR2E11C002B	9.26E-04
CC- RS-60_U2	4/4, P4SR2E11C002C P4SR2E11C002A P4SR2E11C002D P4SR2E11C002B	1.71E-03

2.1.2 Quantification Truncation

To address limitations with each hazard requiring a different truncation level, each hazard was calculated individually and then combined into a single aggregated results file. To generate both the base and case risk, each hazard was quantified at the truncation levels below, to ensure that the basic events for the 2D RHR pump were present.
Internal Events CDF – 1E-12
Internal Events LERF – 1E-12

Internal Flooding CDF – 1E-12
Internal Flooding LERF – 1E-14
Internal Fire CDF – 1E-11
Internal Fire LERF – 1E-12
Seismic CDF – 1E-11
Seismic LERF – 1E-11

2.1.3 Calculation Approach

The proposed technical specification change involves unavailability of the 2D RHR Pump. The revised CDF and LERF values for the CT configurations are obtained by re-quantifying the base PRA model with all of the identified events set as shown in Table 2.1-2.

The evaluation of ICCDP and ICLERP for this condition is determined as shown below: The ICCDP associated with RHR Pump 2D OOS for a new CT is given by:

$$ICCDP_{2d} = (CDF_{2d} - CDF_{BASE}) \times CT_{NEW} \qquad [\textit{Eq. 2-1}]$$

where

CDF_{2d} = the annual average CDF calculated with RHR Pump 2D OOS and other currently OOS equipment assuming the configuration listed in Table 2.1-2 (all quantified hazards)

CDF_{BASE} = baseline annual average CDF with average unavailability for all equipment. This is the CDF result of the baseline PRA (all quantified hazards). Currently OOS equipment other than the RHR Pump 2d was not included in the base case values.

CT_{NEW} = the new extended CT (in units of years)

Note: ICCDP is a dimensionless probability.

Risk significance relative to ICLERP is determined using equations of the same form as noted above for ICCDP.

Since this evaluation is for a one-time Tech Spec CT allowance, the ICCDP and ICLERP are the only meaningful metrics as there is no permanent change in plant risk after this one-time CT extension.

2.1.4 Common Cause Adjustments

RG 1.177 contains specific directions on adjusting the CCF events in a model due to a failed component. The RHR pump failure events are in common cause groups of four for both failure on demand and failure to run. Thus, the common cause events that contain pump D have to be changed to the alpha values for that combination. The common cause basic event values are calculated in Cafta using the formula Qt*CCFM*1 for failure to start and Qt*CCFM*24 hours for failure to run. Qt times the demands or run hours is the overall random failure to start or failure to run probability. CCFM is the common cause multiplier based on the alpha method. The values were obtained from the Hatch data calculation H-RIE-IEIF-U00-007 attachment 4.

For RHR pump failure to start

CCFM, 2 of 4 FTS = 8.92E-03. The three basic events for 2/4 containing pump 2D are increased from 8.232E-06 to 8.92E-03.

CCFM, 3 of 4 FTS = 1.05E-03. The three basic events for 3/4 containing pump 2D are increased from 9.63E-07 to 1.05E-03.

CCFM, 4 of 4 FTS = 1.67E-03. The basic event for 4/4 pumps was increased from 1.54E-06 to 1.67E-03.

For RHR pump failure to run

CCFM, 2 of 4 FTR = 5.10E-03. The three basic events for 2/4 containing pump 2D are increased from 1.56E-06 to 5.10E-03.

CCFM, 3 of 4 FTR = 9.26E-04. The three basic events for 3/4 containing pump 2D are increased from 2.85E-06 to 9.26E-04.

CCFM, 4 of 4 FTR = 1.71E-03. The basic event for 4/4 pumps was increased from 5.25E-07 to 1.71E-03.

2.2 OTMHM Quantification

The relevant inputs from the PRA models to Equation 2-1 (and the equivalent for LERF) are shown in Table 2.2-1 below.

Table 2.2-1

**OTMHM RISK ASSESSMENT
INPUT PARAMETERS AND
RESULTS FOR UNIT 2**

Input Parameter	Value
CDF_{BASE}	6.10E-05
CDF_{2D}	8.59E-05
Delta CDF	2.49E-05
ICCDP for 15 day LCO	**1.02E-06**
$LERF_{BASE}$	3.76E-03
$LERF_{2D}$	3.80E-06
Delta LERF	4.00E-09
ICLERP for 15 day LCO	**1.6E-09**

<u>Compensatory Measures Discussion</u>

Risk insights from this configuration were examined by comparing the change in Birnbaum values between the base and configuration specific importance rankings. The events and components that become more important are associated with the redundant ECCS pumps and the containment hardened vent system, which is an alternate heat sink if RHR is not available.

RG 1.177 requires a Tier 2 examination of other components that, in combination with the component already out of service, could result in a risk significant configuration. For the RHR 2D pump, the components below had risk increases greater than a factor of three.

Components that become more significant due to the configuration:

Item	Base RAW	Case RAW	Factor Increase
2P52F1171	2.17	17.82556	8.219726
2P52F1187	2.17	17.82556	8.219726
2T48F082	2.76	20.29294	7.364984
2T48F081	2.78	20.39179	7.337937
2R25S067	4.67	21.84714	4.680566
2R25S006	3.48	13.86468	3.988584
2T48D346	14.33	46.91086	3.273964

The above components do not have maintenance performed on them while the plant is at power, so no additional restrictions are needed.

RG 1.177 also requires a Tier 3 examination of the (a)(4) Maintenance Rule configuration risk impact. The 2D RHR pump was input into the on-line configuration risk management (CRM) program as a “what-if” evaluation. The Hatch CRM program calculates both the instantaneous and integrated risk and CRM risk levels are based on integrated risk levels. The components already out of service prior to the RHR pump discovery were left out of service for this evaluation to ensure the calculation is conservative. With the ‘B’ pump out of service, the increase in risk is minimal. The CRM program uses the same hazard models that were used for this evaluation, and since the a(4) process evaluates planned work as well as current configurations, it will identify any potential high-risk conditions during the extended CT. The a(4) process of assessing and managing that risk will adequately control the evolution and risk management actions will be generated as necessary.

2.3 EXTERNAL EVENTS

2.3.1 Assessment of Relevant Hazard Groups

The purpose of this portion of the assessment is to evaluate the spectrum of external event challenges to determine which external event hazards should be explicitly addressed as part of the Condition 3.7.1.A extension risk assessment.

Internal events, internal flooding, internal fires, and seismic are quantitatively addressed as described in the previous sections.

The impact due to seismic and other hazard groups are addressed here. It is noted that it is unnecessary to evaluate the low-power and shutdown contribution to the base CDF and LERF since the change being proposed involves performance of the repair while at-power. It should be noted that use of the RHR Pumps is required for shutdown cooling and failure of a pump necessitates entry into other LCOs during shutdown.

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2.3.3 Other External Hazards Evaluation and Conclusions

A plant-specific evaluation of an extensive set of other external hazards (including high winds and external flooding) was performed in SNC calculation H-RIE-OEE-U00 [Ref. 7]. The results have been previously submitted 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 using the criteria in ASME PRA Standard RA-Sa-2009 and concluded that all other external hazards can be screened from applicability at Hatch. Therefore, there is no significant other external hazards risk contribution for this application.

2.4 RESULTS COMPARISON TO ACCEPTANCE GUIDELINES

The values from Table 2.2-1 show that the delta CDF is more limiting. The time to reach 1E-06 ICCDP would be approximately 14.6 days total. The time to reach 1E-05 ICCDP would be >1 yr. Using equation 2-2 with the limiting delta CDF, the ICCDP for any given number of days can be determined.

The results indicate a one-time extension up to 15 days would not exceed the ICCDP and ICLERP risk limits. Additional compensatory measures would potentially reduce risk further, such as protected equipment and abstaining from entry into the diesel generator 14-day LCO or other activities that impact diesel generator availability. The additional compensatory measures are not accounted for in the quantification.

2.5 UNCERTAINTY ASSESSMENT

The purpose of this section is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for Condition 3.5.1.A CT extension assessment. The baseline internal events PRA, internal flooding PRA, and fire PRA (FPRA) models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to Condition 3.5.1.A CT extension assessment, discuss the results, and to provide dispositions. As discussed in Section 1.5, no key assumptions or sources of uncertainty were identified that uniquely impact this application.

3.0 TECHNICAL ADEQUACY OF PRA MODEL

This section provides information on the technical adequacy of the Hatch Nuclear Plant Probabilistic Risk Assessment (PRA) models. 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, specifically in support of the requested extended CT for TS Condition 3.5.1.A. The OTMHM is comprised of various hazards PRA models that can be quantified simultaneously or individually. Each hazard (internal events, internal flooding, fire, and seismic) has been peer reviewed. The most up-to-date assessments of PRA technical adequacy (including peer review status, F&O closure status, scope, fidelity, capability, and maintenance/update practices) was provided to the NRC previously for the Hatch 50.69 LAR (ADAMS Accession Number ML18158A583) and subsequent RAI responses (ML19197A097); and also the NFPA-805 LAR (ML18096A955) and subsequent RAI responses (ML19280C812). Additionally, those submittals contain the most up-to-date description of the other external hazards assessment.

4.0 SUMMARY AND CONCLUSIONS

This analysis evaluates the acceptability, from a risk perspective, of a change to the Hatch Unit 2 TS Condition 3.5.1.A for a one-time increase of the CT from 7 days to 15 days when the RHR Pump 2D is inoperable.

The analysis examines a range of risk contributors including internal events, internal flooding, fire, seismic, shutdown risk and other external hazards. The configuration was quantified using the Phoenix OTMHM model and compared to the base risk to obtain delta CDF and LERF values.

4.1 PRA QUALITY

The PRA quality has been assessed and determined to be adequate for this risk application, and the PRA technical adequacy has also been addressed in recent NRC submittals.

To summarize,

- Scope – Hatch PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA has the necessary scope to appropriately assess the pertinent risk contributors.
- Fidelity –The Hatch PRA models are the most recent evaluation of the risk profile. The PRA reflects the as-built, as-operated plant, with the exception of previously noted items.
- Standards – The PRA has been reviewed against the ASME/ANS PRA Standard and the PRA elements are shown to have the necessary attributes to assess risk for this application.
- Peer Review - The PRA has received a peer review. Based on addressing the peer review results and subsequent gap analyses to the current standards, the PRA is found to have the necessary attributes to assess risk for this application.
- Appropriate Quality – The PRA quality is found to be appropriate to assess risk for this application.

4.2 QUANTITATIVE RESULTS VS. ACCEPTANCE GUIDELINES

This analysis demonstrates with reasonable assurance that the proposed TS change is within the current risk acceptance guidelines in RG 1.177 for one- time changes.

4.3 CONCLUSIONS

This analysis demonstrates the acceptability, from a risk perspective, of a change to the Hatch TS Condition 3.5.1.A to increase the CT from 7 days to 15 days when the RHR Pump 2Dis unavailable.

A PRA technical adequacy evaluation was also performed consistent with the requirements of ASME/ANS PRA Standard and RG 1.200, Revision 2. Additionally, a review of model uncertainty and outstanding changes was performed with this application. None of the identified sources of uncertainty were significant enough to change the conclusions from the risk assessment results presented here.

5.0 REFERENCES

[1] Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities,” Revision 3, December 2020.

[2] Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk- Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018.

[3] Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 2, January 2021.

[4] ASME/ANS RA- Sa-2009, February 2009. “Addenda to RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,”

[5] H-RIE-PHOENIX-U01, “One Top Model for PHOENIX Configuration Risk Management Program.”

[6] H-RIE-OEE-U00 – “Hatch Other External Events Screening”

[7] PRA-BC-H-21-003 – “RHR Pump 2D Emergent Technical Specification Change”