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

Edwin I. Hatch Nuclear Plant Unit 1
Emergency License Amendment Request for Technical Specification 3.7.2
Regarding One-Time Extension of Completion Time for
Plant Service Water (PSW) Pump Inoperable

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 1 renewed facility operating license DPR-57.

The proposed amendment would revise TS 3.7.2, "Plant Service Water (PSW) System and Ultimate Heat Sink," Limiting Condition for Operation (LCO) 3.7.2, Condition A, "One PSW pump inoperable," to allow a one-time increase in the Completion Time from 30 days to 45 days. The increased Completion Time would expire on October 10, 2021 at 1620 eastern daylight time (EDT).

The one-time only change allows for continued repair and testing activities on the HNP Unit 1C PSW pump. The expiration date for the proposed allowance is based on the current 30-day Completion Time expiration at 1620 EDT, September 25, 2021, plus the requested additional 15 days (45 days total).

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

SNC requests approval of the proposed license amendment as soon as possible and no later than September 24, 2021 based on emergent circumstances at HNP Unit 1 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 a marked-up TS page and revised TS page, 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 previously 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 Ryan Joyce at 205-992-6468.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 21st day of September 2021.

Respectfully submitted,

Cheryl A. Gayheart
Regulatory Affairs Director

CAG/tle

Enclosure: Description and Assessment of the Proposed Change

Attachments:

1. HNP Unit 1 Technical Specification Marked-up Page
2. HNP Unit 1 Revised Technical Specification Page
3. HNP Unit 1 Technical Specification Bases Marked-up Page (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 1

Emergency License Amendment Request for Technical Specification 3.7.2
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Plant Service Water (PSW) Pump Inoperable

Enclosure

Description and Assessment of the Proposed Change

1. Summary Description

The proposed amendment to Hatch Nuclear Plant (HNP) Unit 1 renewed facility operating license DPR-57 would revise Technical Specification (TS) 3.7.2, "Plant Service Water (PSW) System," Limiting Condition for Operation (LCO) 3.7.2, Condition A, "One PSW pump inoperable," to allow a one-time increase in the Completion Time from 30 days to 45 days. The allowance would only be applicable while the compensatory measures described in Section 3.3 of this application and as affirmed in the NRC's safety evaluation are implemented, would only apply to PSW pump 1C, and would expire on October 10, 2021, at 1620 eastern daylight time (EDT).

On August 26, 2021, at 1620 EDT, HNP Unit 1 PSW pump C was declared inoperable due to excessive pump vibration. As described in Section 2.1 of this application, the original scope of work to restore the pump to Operable was to replace the pump which was anticipated to take eight to nine days. However, as the result of various discoveries during initial pump replacement activities, repair work was more complex and involved additional personnel high risk activities than originally anticipated. These discoveries led to repair work not being completed until September 16, 2021 and are described in detail in Section 2.1. During the post-maintenance operability test run on September 16, 2021, a loud high-pitched noise was heard from the pump, at which time the pump was secured. As discussed in Section 2.1, additional discovery from September 16 to September 18, 2021 has required HNP to replace the pump and refurbish the motor off-site at a specialty vendor.

The comprehensive repair work has been time-consuming; however, Southern Nuclear Operating Company (SNC) has demonstrated due diligence by safely performing testing and maintenance activities without delay. SNC now expects that repair and testing will extend past the 30-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 1 PSW pump C to Operable status. To provide allowance for additional time to complete repairs, SNC is requesting a one-time Completion Time extension from 30 days to 45 days, applicable to the 1C PSW pump only.

2. Detailed Description

2.1 <u>Emergency Circumstances</u>

Why the Condition Occurred:

On August 26, 2021 at 1620 EDT, HNP Unit 1 PSW pump 1C was declared inoperable due to excessive pump vibration. The emergency circumstances result from the unforeseen failure of the HNP Unit 1 PSW pump 1C during its post-maintenance operability test on September 16, 2021, nine days before the expiration of the required Completion Time for Condition A for TS 3.7.2. The events that led to these emergency circumstances and the facts surrounding the initial event (Initial Discovery) and post-maintenance operability test (Additional Discovery) are documented below.

SNC has been performing test and repair activities, and the pump and motor vendors are engaged with the investigation and repair activities. The required Completion Time for Condition A for TS 3.7.2 of 30 days is currently applicable and will expire on September 25, 2021, at 1620 EDT. The current repair and replacement activities along with the identified contingencies associated with repairs are unlikely to be complete during the remaining allowed LCO time. Neither a routine nor an exigent amendment can be processed prior to September 25, 2021, at 1620 EDT.

It is noted that at the time of Initial Discovery up to Additional Discovery, SNC had no reasonable expectation that the TS Completion Time would not be met.

Initial Discovery

During normal operator rounds on August 26, 2021, the 1C PSW pump was observed to have excessive vibration, and in turn, Operations took action to secure the pump at which time it was declared inoperable. The station took action to begin troubleshooting the event and visual inspections identified:

- All four motor to pump discharge fasteners were loose and could be turned by hand;
- One of the pump discharge head to floor fasteners was loose; and
- A significant gap existed between the seal box drive collar and gland plate assembly.

Due to the visual evidence found, the motor was uncoupled from the pump for further inspection and in preparation for an uncoupled motor run. The purpose of this uncoupled run was to ascertain the health of the motor following the initial event. Troubleshooting also included collecting vibration data during the uncoupled run, and motor integrity testing of the motor.

An uncoupled motor run was performed on August 26, 2021. The highest vibration during the uncoupled run was within the acceptable range. Based on results of the uncoupled run and troubleshooting performed, there was no indication of motor damage resulting from the excessive vibration. It was determined at that time that the appropriate action was to replace the pump only. The original schedule for pump replacement had the pump Operability restored by roughly 21 days prior to the expiration of the TS Completion Time.

On August 30, 2021, during pump replacement activities, it was discovered that the pump shaft had broken inside of the coupling where the bottom pump column bolts to the suction head. The bolts at the suction head to column flange connection were also discovered to be loose, broken, or missing. At that time, it was also identified that the suction head was no longer connected to the pump column and remained submerged in the intake suction pit. A foreign material (FM) retrieval plan and evaluation was requested and preparations for diving activities began.

On September 2, 2021, divers in the PSW suction pit identified the pump suction head was directly below the 1C PSW pump location, standing vertically in its original position. The pump suction head was removed from the suction pit on September 3, 2021; however, the remaining FM (nuts, bolts and washers) had not been accounted for at that time. Due to limited visibility in the water and proximity of other in-service pumps, a time-out was called to identify an alternate method for foreign material retrieval while pump replacement activities continued in parallel. A loose parts evaluation was initiated for FM in the suction pit. The loose parts evaluation was developed by industry experts to ensure flow characteristics of the pump and FM design and weight were considered.

On September 4, 2021, it was discovered that the upper seismic restraint required adjustment to meet pump column alignment requirements. Modification of the upper seismic restraint was required along with the addition of shims to ensure compliance with design specifications and proper alignment.

On September 7, 2021, it was determined that the outer strap of the seismic restraint also required adjustment. The strap was removed and inspected by site Quality Control. No abnormal conditions were identified.

On September 15, 2021, the loose parts evaluation was completed. The evaluation concluded that parts directly below the suction bell of the pump were susceptible to being ingested by the pump. This information was given to Maintenance to ensure vacuuming activities removed all potential FM below the suction bell. Additionally, Quality Control performed a VT-3 inspection of the lower seismic support strap which identified the need to modify the seismic support and add shims. Shims were welded in to get the required spacing for the lower seismic support.

On September 16, 2021, following adjustments to seismic restraints and installation of the newly assembled pump and existing motor, a post-maintenance test was performed. The pump was secured after approximately 15 seconds of run time due to a loud high-pitched noise, at which time the pump troubleshooting was initiated to identify the cause.

<u>Additional Discovery</u>

After the 1C PSW pump and motor were secured, maintenance troubleshooting was initiated to identify the cause of the loud high-pitched noise. Motor oil sampling was performed and identified increased levels of particulate. Wear particle counts were within acceptable ranges per industry guidance, but were unexpected for fresh oil with low run time. Motor oil was replaced during motor installation following the pump rebuild. Due to the elevated wear particle count on fresh oil and the motor only running approximately 15 seconds, the health of the motor was questioned. The decision was made to remove the motor and send it offsite for vendor investigation and refurbishment.

On September 17, 2021, vendor technicians identified that the lower guide bearing caps were axially misaligned by 0.030 inch which led to heavy wear, or wiping, of the lower guide bearing during the post-maintenance test. When the motor was uncoupled, the pump did not release and drop down as expected. The pump lift is ½-inch, but no drop occurred after motor uncoupling. While attempting to remove the mechanical seal components, it appeared that electrical arcing had occurred on the mechanical seal and coupling components.

On September 18, 2021, the motor refurbishment vendor identified that the thrust bearing had been dislodged from its housing. The vendor reported that the pump and motor shaft total upthrust was between 1.5 and 2.0 inches. Based on vendor documentation, expected total axial endplay or normal upthrust would be less than 0.020 inch. Investigation by the vendor indicates the most probable cause of damage to the motor was a significant upthrust event which most likely occurred during the event on August 26, 2021 when the pump was damaged. Motor damage was undetected through initial troubleshooting of the motor due to being unloaded during the uncoupled run. Continued troubleshooting on the pump by HNP identified damage to the newly installed pump due to electrical arcing, the cause of which is still under investigation. To prevent future damage due to arcing, HNP will ensure no welding activities occur while the pump is installed. In addition, the motor is being refurbished to eliminate internal faults as a possibility.

Next Steps

With the current uncertainties surrounding damage to the pump and motor, on September 18, 2021 HNP made the decision to replace the pump and refurbish the motor. This will remove any new discovery points associated with pump or motor damage while ensuring a healthy and reliable 1C PSW pump. This is the point in which HNP determined they would be unable to meet the current LCO time.

Why this Situation Could Not be Avoided:

Based on the investigation performed following the initial event on August 26, 2021, HNP had evidence suggesting that the motor was not damaged due to excessive pump vibrations. As described above, this evidence included an uncoupled run, vibration analysis, and motor integrity testing. Further inspection of the motor would have required disassembly and inspection by an offsite vendor. HNP at that time made the decision to replace the pump with a return to service date of September 4, 2021 providing sufficient margin to the expiration of the TS Completion Time. Due to the multiple points of discovery with the pump, foreign material, and seismic restraints, the normal replacement duration was delayed but still completed nine days prior to the current TS Completion Time.

During the post-maintenance test on September 16, 2021, the 1C PSW pump and motor were shut down after approximately 15 seconds of run time due to a loud high-pitched noise. Following the post-maintenance test the motor was determined to be damaged. Based on vendor investigation it was determined that the motor was subjected to an up-thrusting event which caused damage to the motor and pump. The damage to the motor following the post maintenance run was not evident until disassembly and inspection by the offsite vendor. Investigation of the pump also identified wear and electrical arcing on the pump shaft and mechanical seal components. Due to the cause of the arcing currently being unknown, the decision was made to replace the pump assembly to ensure health and reliability of the pump and motor.

Need for an Emergency Amendment pursuant to 10 CFR 50.91(a)(5)

Immediately following the post-maintenance test run on September 16, 2021, SNC began investigating the cause of the post-maintenance test anomaly. Although it was unknown at that time whether the TS allowed Completion Time would be challenged, SNC shortly thereafter began preparation of various inputs (e.g., required risk assessments) needed for a contingency license amendment request (LAR). On September 18, 2021, seven days prior to the expiration of the applicable TS Completion

Time, it was concluded that both pump and motor replacement would be required. Without delay, work commenced to develop the schedule and secure the needed technical expertise to complete these replacements.

The determination of a requested extension to the TS Completion Time required a thorough review of the repair schedule in order to assure: (a) the request was reasonable (i.e., not excessive and unnecessarily long); and (b) there is high probability that the 1C PSW pump would be restored to Operable status within the requested Completion Time. In addition, to provide assurance that the incurred risk to the plant is acceptable and that high-risk configurations will be avoided during the extended TS Completion Time, a risk-analysis meeting the requirements of Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," Revision 2 (Reference 1) was performed. This analysis is summarized in Attachment 4 of this letter. This risk analysis identified compensatory measures that are required to be implemented during the duration of the proposed extended Completion Time.

SNC has been diligent in the preparation of this LAR. Processing an exigent LAR pursuant to 10 CFR 50.91(a)(6) requires reasonable notice to the public in the areas surrounding the facility of the proposed LAR and of the determination of the no significant hazards consideration as described in 10 CFR 50.91(a)(2). However, based on both: (a) the timing of the realization that a LAR is needed; and (b) the development of required inputs to prepare this LAR (e.g., detailed repair schedule), submittal of an exigent LAR was not feasible. Therefore, an emergency amendment pursuant to 10 CFR 50.91(a)(5) is necessary.

To allow adequate time for the preparation of a TS required shutdown (in the event this amendment is denied), SNC is requesting NRC approval by September 24, 2021 (i.e., six days after it was determined an emergency TS change was required). Therefore, an exigent TS change per 10 CFR 50.91(a)(6) was not feasible once the full extent of condition of the pump and motor was understood.

Operational Experience (OE) Review

SNC has reviewed recent applicable operating experience within our fleet and the industry regarding vertical pumps. The most relevant OE came from an HNP Service Water pump that experienced elevated vibrations due to misalignment of the pump column. This misalignment was caused by lower seismic restraints that were not aligned with the column, which in turn caused misalignment of the shaft and increased vibration. These restraints were verified to not cause misalignment of the 1C PSW pump during initial discovery.

Additional OE from SNC's Farley Nuclear Plant noted damage to a vertical pump due to a wiring issue. As discussed above, the indication of electrical arcing on the 1C PSW pump is still being investigated at this time. In addition to these specific SNC OE reviews, various industry and fleet subject matter experts have been consulted throughout this system outage to ensure that broader knowledge of potential issues is incorporated into the review and return to service of the 1C PSW pump. The inclusion of the feedback from subject matter experts, as well as validation of other fleet OE noted above, is being utilized to ensure a full understanding and resolution of any anomalies prior to return to service.

In addition, industry and pump vendor operating experience (OE) was reviewed, as well as NRC Information Notices (INs) (e.g., INs 93-68, 94-45, 96-36, and 07-05). This review did not uncover any OE that would suggest a potential common cause mode failure for the 1C PSW pump.

Summary of Emergency Circumstances:

In summary, the emergency circumstances resulted from the unforeseen failure of the HNP Unit 1 PSW pump 1C during post-maintenance testing. The required Completion Time for Condition A for TS 3.7.2 of 30 days is currently applicable and will expire on September 25, 2021 at 1620 EDT. Based on further discovery on and since September 16, 2021, SNC is unlikely to complete repairs of the PSW pump 1C by that time. Neither a routine nor an exigent amendment can be processed prior to September 25, 2021 at 1620 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. Experienced pump and motor vendors are engaged for assistance. Therefore, efforts were made to minimize the likelihood for delays due to job planning or preparation.

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 1 without an approved amendment. If the proposed license amendment is not approved, Unit 1 will be required to enter shutdown TS 3.7.2 Condition E on September 25, 2021 at 1620 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 <u>System Design and Operation</u>

The PSW System is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs), residual heat removal (RHR) pump coolers, and room coolers for Emergency Core Cooling System equipment, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The PSW System also provides cooling to unit components, as required, during normal operation. Upon receipt of a loss of offsite power or loss of coolant accident (LOCA) signal, nonessential loads are automatically isolated, the essential loads are automatically divided between PSW Divisions 1 and 2, and one PSW pump is automatically started in each division.

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

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

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

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

The ability to provide onsite emergency AC power is dependent on the ability of the PSW System to cool the DGs. The long-term cooling capability of the RHR, core spray, and RHR service water pumps is also dependent on the cooling provided by the PSW System. In the analysis presented in Reference 2, only one PSW pump is required for safe shutdown, including RHR Shutdown Cooling System requirements.

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

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

2.3 Current Technical Specification Requirements

Currently, TS 3.7.2, Condition A, requires restoration of an inoperable PSW pump to operable status within 30 days.

2.4 Reason for the Proposed Change

Despite diligent and prudent efforts, SNC has been unable to return the PSW pump 1C to operable service and now expects it will be unable to do so by expiration of the Completion Time for TS 3.7.2, Condition A (i.e., September 25, 2021 at 1620 EDT) and would then be required to place HNP Unit 1 in Mode 3. Approval of the proposed change would allow SNC to continue repair and replacement activities as necessary and without undue risk as demonstrated in Attachment 4 of this license amendment request.

The table below summarizes the maintenance tasks to be performed for the 1C PSW pump and the current expected completion date of each task. SNC will work to restore the 1C PSW pump to Operable status as soon as reasonably and safely achievable, regardless of the additional Completion Time duration.

Task No.	Project Task	Expected Task Completion Date
*1	Remove Motor and Send for Repair	9/17/2021
2	Remove Pump	9/21/2021
3	Receive Refurbished Motor	9/23/2021
4	Install New Pump	10/4/2021
5	Install Refurbished Motor	10/5/2021
6	Restore Pump to Operable	10/6/2021

*Task already performed.

While SNC believes the above schedule is realistic, there are inherent uncertainties and unknowns associated with major pump and motor maintenance. To account for these uncertainties and unknowns, SNC is requesting an additional four days beyond the current maintenance schedule to restore the 1C PSW pump to Operable status. Thus, as supported by this LAR, SNC requests until October 10, 2021, at 1620 hours EDT (i.e., a total 45-day Completion Time), to complete this work.

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 Description of the Proposed Change

The following revision is proposed to TS 3.7.2 Condition A (added text in *italics*):

-------------------------------*NOTE*-------------------------------

A Completion Time of 45 days is permitted for Pump 1C while the compensatory measures described in Section 3.3 of SNC letter NL-21-0852 dated September 21, 2021 are implemented. This allowance expires at 1620 EDT on October 10, 2021.

The one-time only change allows for continued repair and testing activities in a prudent fashion. The expiration date of October 10, 2021 at 1620 EDT is based on the current 30-day Completion Time expiration at 1620 EDT, September 25, 2021, plus the requested additional 15 days (45 days total). The allowance allows SNC to safely address any additional unforeseen circumstances, such as weather conditions, or discoveries, such as the need to order new parts or to contract with a specialty vendor.

The allowance would only apply to the 1C PSW pump and only as long as the compensatory measures described in Section 3.3 of this application are implemented.

3. Technical Evaluation

3.1 Defense-in-Depth

During the extended Completion Time, the PSW System will remain within the limits of the Technical Specifications. Should an event occur requiring the PSW System and the UHS (i.e., the Altamaha River), the remaining PSW pumps are capable of performing the safety function of providing cooling water.

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 PSW System.
- The PSW system has sufficient capacity to function for design basis events while in Condition A. The UFSAR acceptance criteria for the design events will be met should such an event occur during the time that the 1C PSW pump is out of service. It is noted that in the analysis presented in Reference 2, only one PSW pump is required for safe shutdown, including RHR Shutdown Cooling System requirements.

3.3 Compensatory Measures

The following compensatory measures are required during the extended Completion Time.

- The following equipment is protected as required by SNC Procedure NMP-OS-010-002 (Reference 3) for 1C PSW pump out-of-service:
  - 1A PSW Pump
  - 1E 4160V Frame 3 (power supply to 1A PSW Pump)
  - 1A PSW Pump Control Switch
- Travelling water screen 1B will be placed in RUN if the 1A screen is taken out of service.
- HNP Operations (each shift) will review the abnormal procedure for loss of PSW, SNC Procedure 34AB-P41-001-01 (Reference 4).
- High Pressure Coolant Injection (HPCI) will be protected with work limited to TS required surveillances only.

- Reactor Core Isolation Cooling (RCIC) will be protected with work limited to TS required surveillances only.
- No maintenance will be performed on 1T48F081 or 1T48F082, the Containment Hardened Vent path.
- The 1B diesel generator and the Standby Service Water (SSW) pump will be protected, and work limited to TS required surveillances only.
- All three Unit 1 startup transformers and their associated 230KV breakers will be protected.
- No preventive maintenance will be performed on the FLEX pumps to ensure their availability during the extended Completion Time.

3.4 Maintenance Rule Control

The PSW pumps are included under the HNP Maintenance Rule Program. The PSW 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 PSW 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.

Pump 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 1 TS 3.7.2, "Plant Service Water (PSW) System and Ultimate Heat Sink (UHS)," Condition A, to allow a one-time increase in the Completion Time from 30 days to 45 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 5).

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 6), and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," Revision 2 (Reference 1).

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 design of the PSW System satisfies 10 CFR 50.36, "Technical Specifications," paragraph (c)(2)(ii), Criterion 3, which states the following:

> "(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
>
> …
>
> *Criterion 3*. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

The PSW System is described in the HNP Unit 1 UFSAR Section 10.7 (Reference 2).

The proposed amendment does not delete requirements associated with the PSW System and LCO 3.7.2 continues to maintain requirements associated with structures, systems, and components that are part of the primary success path and actuate to mitigate the related design basis accidents and transients. The proposed amendment does not alter the remedial actions or shutdown requirements required by 10 CFR 50.36(c)(2)(i). The proposed changes do not affect compliance with this regulation.

Following implementation of the proposed change, HNP Unit 1 will remain in compliance with applicable Atomic Energy Commission (AEC) design criteria as described in the HNP Unit 1 UFSAR (Reference 7).

4.2 Precedent

There are no identified precedents significantly relevant to this amendment.

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 1 renewed facility operating license DPR-57. The proposed amendment would revise Condition A, of LCO 3.7.2, of Technical Specification (TS) 3.7.2, "Plant Service Water (PSW) System and Ultimate Heat Sink (UHS)," to extend the Completion Time from 30 days to 45 days for the 1C PSW pump only. This proposed allowance would expire on October 10, 2021 at

1620 eastern daylight time (EDT) and be effective only while the compensatory measures described in Section 3.3 of this application are implemented.

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

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

2. Does the proposed amendment 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 PSW System will remain capable of providing the required cooling to systems responsible for mitigating the consequences of a design basis event such as a LOCA.

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

Based on the above, 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. Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications*, Revision 2, January 2021
2. HNP-1-FSAR, Edwin I. Hatch Nuclear Plant Final Safety Analysis Report, Revision 38, September 2020, Section 10.7, “Plant Service Water System”
3. SNC Procedure NMP-OS-010-002, *Hatch Protected Equipment Logs*, Version 11.1, Effective March 4, 2020
4. SNC Procedure 34AB-P41-001-01, *Loss of Plant Service Water*, Version 11.8, Effective May 26, 2021
5. Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2, March 2009

6. 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
7. HNP-1-FSAR, Edwin I. Hatch Nuclear Plant Final Safety Analysis Report, Revision 38, September 2020, Section F.5, “Evaluation with Respect to 1971 General Design Criteria”

Edwin I. Hatch Nuclear Plant Unit 1

Emergency License Amendment Request for Technical Specification 3.7.2 Regarding One-Time Extension of Completion Time for Plant Service Water (PSW) Pump Inoperable

Attachment 1

HNP Unit 1 Technical Specification Marked-up Page

3.7 PLANT SYSTEMS

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

LCO 3.7.2 Two PSW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PSW pump inoperable.	A.1 Restore PSW pump to OPERABLE status.	------------NOTE------------ A Completion Time of 45 days is permitted for Pump 1C while the compensatory measures described in Section 3.3 of SNC letter NL-21-0852 dated September 21, 2021 are implemented. This allowance expires at 1620 EDT on October 10, 2021. -------------------------------- 30 days
B. One PSW turbine building isolation valve inoperable.	B.1 Restore PSW turbine building isolation valve to OPERABLE status.	30 days
C. One PSW pump in each subsystem inoperable.	C.1 Restore one PSW pump to OPERABLE status.	7 days
D. One PSW turbine building isolation valve in each subsystem inoperable.	D.1 Restore one PSW turbine building isolation valve to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 ------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours

(continued)

Edwin I. Hatch Nuclear Plant Unit 1

Emergency License Amendment Request for Technical Specification 3.7.2
Regarding One-Time Extension of Completion Time for
Plant Service Water (PSW) Pump Inoperable

Attachment 2

HNP Unit 1 Revised Technical Specification Page

3.7 PLANT SYSTEMS

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

LCO 3.7.2 Two PSW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PSW pump inoperable.	A.1 Restore PSW pump to OPERABLE status.	------------NOTE------------ A Completion Time of 45 days is permitted for Pump 1C while the compensatory measures described in Section 3.3 of SNC letter NL-21-0852 dated September 21, 2021 are implemented. This allowance expires at 1620 EDT on October 10, 2021. -------------------------------- 30 days
B. One PSW turbine building isolation valve inoperable.	B.1 Restore PSW turbine building isolation valve to OPERABLE status.	30 days
C. One PSW pump in each subsystem inoperable.	C.1 Restore one PSW pump to OPERABLE status.	7 days
D. One PSW turbine building isolation valve in each subsystem inoperable.	D.1 Restore one PSW turbine building isolation valve to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 ------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours

(continued)

Edwin I. Hatch Nuclear Plant Unit 1

Emergency License Amendment Request for Technical Specification 3.7.2
Regarding One-Time Extension of Completion Time for
Plant Service Water (PSW) Pump Inoperable

Attachment 3

HNP Unit 1 Technical Specification Bases Marked-up Page (information only)

BASES

APPLICABILITY (continued)

The LCO for the PSW System and UHS is not applicable in MODES 4 and 5, and defueled. However, portions of the PSW System and UHS may be required to perform necessary support functions for OPERABILITY of the supported systems. Thus, the LCOs of the individual systems, which require portions of the PSW System and the UHS to be functional to support individual system OPERABILITY, will govern PSW System and UHS requirements during operation in MODES 4 and 5 and defueled.

ACTIONS

A.1

With one PSW pump inoperable, the inoperable pump must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE PSW pumps (even allowing for an additional single failure) are adequate to perform the PSW heat removal function; however, the overall reliability is reduced. The 30 day Completion Time is based on the remaining PSW heat removal capability to accommodate additional single failures, and the low probability of an event occurring during this time period.

The Completion Time is modified by a Note indicating allowance to extend the Completion Time from 30 days to 45 days until 1620 EST on October 10, 2021 for PSW pump 1C while the compensatory measures described in Section 3.3 of SNC letter NL-21-0852 dated September 21, 2021 are implemented. This change was approved by the NRC in September 2021 to allow prudent time to repair and test the pump.

B.1

With one PSW turbine building isolation valve inoperable, the inoperable valve must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE PSW turbine building isolation valve in the subsystem is adequate to isolate the non-essential loads, and, even allowing for an additional single failure, the other PSW subsystem is adequate to perform the PSW heat removal function; however, the overall reliability is reduced. The 30 day Completion Time is based on the remaining PSW heat removal capability to accommodate additional single failures, and the low probability of an event occurring during this time period.

(continued)

Edwin I. Hatch Nuclear Plant Unit 1

Emergency License Amendment Request for Technical Specification 3.7.2
Regarding One-Time Extension of Completion Time for
Plant Service Water (PSW) Pump Inoperable

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 3.7.2.A from 30 days to 45 days for Unit 1 in order to allow for repair of the 1C Plant Service Water (PSW) 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 2

Regulatory Guide 1.200, Revision 2 [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 2 endorses Addendum A of the ASME/ANS PRA Standard [Ref. 4] as clarified in Appendix A of RG 1.200, Revision 2.

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 permanent 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 permanent 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 temporary 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 2 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. Loss of PSW is a modeled special initiating event and logic for this is in the model.
- Internal Flooding (IF) – The Hatch PRA model used for this analysis includes flooding scenarios.
- Low Power Operation – The intent is for the unit to remain at power during the completion time. PSW provides motor cooling for RHR Service Water pumps during shutdown. Since RHRSW is used for shutdown cooling, and RHRSW is cooled by PSW, there is some risk involved with going into lower modes; however, that is not quantified or discussed any further in this assessment. As described below, since the shutdown success criteria only requires one pump, and three are still operable, this is a small increase only.
- Shutdown / Refueling – Hatch does not have a shutdown PRA model, but instead relies upon NUMARC 91-06 deterministic methodology to assess defense-in-depth of key safety functions. PSW is not measured directly but is considered a support system for the key safety functions.
- Internal Fires – The Hatch PRA model used for this analysis contains an as-built, as-operated Fire PRA model. (Note that Internal Fires are often included in the list of “external events” evaluated by a PRA; that is the case for Hatch as well.)
- Seismic - The Hatch PRA model used for this analysis includes a Seismic PRA.
- Other External Events - Other external event risks (including external flooding and high winds) were assessed in the Hatch Other External Events Screening calculation [Ref. 6] and screened from the PRA.

1.5 Hatch PRA MODELS

This section addresses the requirements of Section 3.1 of RG 1.200, Revision 2 [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-U01 [Ref. 5]. This model has the required quantification and support files set up to calculate either zero-maintenance or average maintenance risks. To clarify, it was used to generate average maintenance risk except where adjustments to components are specifically made. 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. Even so, uncertainty associated with some items that are not currently modeled are addressed (Open Phase, Breaker Coordination).

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 addressed. There are two open findings related to internal flooding documentation that do not impact the outcome of this assessment. 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.

PRA Maintenance and Updates

SNC Risk Informed Engineering (RIE) developed a comprehensive PRA model and application maintenance process in response to internal and external assessments and issuance of industry configuration management guidance documents. This process ensures that the applicable PRA models remain as accurate reflections of the as-built and as-operated units. This process delineates the responsibilities and guidelines for updating the PRA models at all operating SNC nuclear generation sites. It defines the process for implementing PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), and for controlling the model and associated computer files. Components of this process include:

- Design change impact reviews are performed by RIE prior to implementation.
- Procedures that can affect PRA modeling or assumptions are reviewed by RIE prior to issue.
- Licensing document changes are reviewed by RIE prior to issue.
- RIE personnel participate in the Maintenance Rule expert panel, Surveillance Frequency Control Program and 10CFR50.69 Independent Decision Making Panels.

SNC Procedure RIE-001 requires that potential impacts to the PRA models be identified and entered in the PRA Model Change log. Each entry in the change log requires an evaluation of the impact of the individual change, as well as an evaluation of the cumulative impact for unincorporated changes. This results in a continuous change tracking process so that the difference between the models and the plant can be quickly determined and evaluated.

In addition to these activities, SNC risk management procedures provide guidance for PRA documentation quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- Requirement to evaluate model changes against the ASME standard definitions of Upgrade and Model Maintenance. Requirement to conduct focused peer review for any changes classified as an Upgrade.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.

- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65 (a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximate two refueling outage cycle; however, longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant.

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 1C PSW 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, Seismic PRA (SPRA) and the Fire PRA (FPRA).

Table 2.1-1
HATCH CDF AND LERF BASE RISK METRICS

Hazard(s)	**Risk (1/yr)**
OTMHM CDF	6.44E-05
OTMHM LERF	4.38E-06

The general configuration for the extended CT is Hatch U1 at-power with the 1C PSW Pump out of service. Additional adjustments are described for each case ran. The risk impact is for Unit 1. 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, using the Configuration Risk Management Program and compensatory measures.

The PRA model was quantified using the base “average test and maintenance” PRA model with the 1C PSW Pump basic events set to TRUE. This included other currently out of service plant equipment. 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 extended completion time was left at their nominal values. Adjustments for common cause factors associated with the 1C PSW pump are also included.

Table 2.1-2
EXTENDED CT CONFIGURATION OUT OF SERVICE REPRESENTATION

Component	Basic Event	Description
1W33E003A	S3PL1W33E003A	upstream traveling water screen [1]
1P41F1500A	MVXC1P41F1500A	U1 wash water to E003A screen [1]
2P41F1500B	MVXC2P41F1500B	U2 wash water to E003A screen [1]
1W32F016A	MVFO1W32F016A	E003A auto wash water valve [1]
1T47B007A	CC-VC-3, CC-VC-13	Failed drywell cooler fan
1P41F397B	HVXC1P41F397B	PSW to heat exchanger B [2]
1P41F398B	HVXC1P41F398B	PSW to heat exchanger B [2]
1P42F397B	HVXC1P42F001B	RBCCW to heat exchanger B [2]
1P42F397B	HVXC1P42F002B	RBCCW to heat exchanger B [2]
Basic events directly associated with PSW pump 1C	CC-PS-1	PSW Pump 1C fails to run for 24 hours
	CC-SW-3	PSW Pump 1C fails to start
	MNUNPS_TRNC	PSW Pump 1C in maintenance
	CC-PS-1________I	PSW Pump 1C fails during the year as part of the Loss of PSW modeled initiating event, set to False for this evaluation
	CC-SW-33	PSW 1C outlet check valve fails to close following PSW 1C fails to run (PSW 1A flow diversion), set to False for this evaluation
	HVXC1P41F009C	cooling water to pump motor inlet
	HVXC1P41F020C	cooling water to pump motor outlet
	HVXC1P41F302C	pump discharge manual isolation valve

Component	Basic Event	Description
1X41C009A	CC-INT-1, FNOS1X41C009A	intake structure vent fan , located in roof plug removed for pump access

1) Components taken OOS to support diving activities in the PSW pit
2) Leaking RBBCW Heat Exchanger Tubes

The PRA model success criteria for PSW pumps is that 1 of 4 pumps is needed to respond to transient events, and 3 of 4 pumps are needed to prevent the Loss of PSW initiating event and keep the unit on-line.

To produce a conservative result, the maintenance terms for components that are protected per the protected train procedure NMP-OS-010 (1P41C001A – Control switch, 4160v breaker and pump.) are left at their nominal values. Since planned maintenance on these components will not be performed while PSW 1C is out of service, this result is conservative.

Per the system operating procedure, with the 1W33E003A upstream traveling water screen out of service, the downstream screen 1W33E001B has been placed in RUN. If 1W33E003A upstream traveling water screen is restored and subsequently taken out of service again during the evolution, the downstream screen will again be place in RUN. The basic events associated with the running screen were not adjusted in this assessment, also adding conservatism
A flag file was used to change the event probabilities of the impacted events. The base model OTMHM flag file was used as a starting point and the above events were added to it.

Common Cause event adjustments due to failed components

RG 1.177 contains specific directions on adjusting the CCF events in a model due to a failed component. The PSW pump failure events are in common cause groups of four for both failure to start and failure to run. Thus, the common cause events that contain pump C must be changed to the alpha values for that combination.

The Phoenix model uses formulas to calculate the values for CCF events. The general form of this formula is Qt * MT * CCFA, where Qt is the total random failure rate, MT is the mission time, generally 1.0 for start failures and 24 for run failures, and CCFA is the common cause adjustment factor. The CCFA factor consists of the alpha factors multiplied by the fraction of combinations it will appear in.

On August 30, 2021, during pump replacement activities in response to the pump failure on August 26th, it was discovered that the pump shaft had broken inside of the coupling where the bottom pump column bolts to the suction head. In addition, the pump suction head was no longer connected to the pump column and remained submerged in the intake suction pit. A foreign material (FM) retrieval plan and evaluation was requested and preparations for diving activities began.

On September 2, 2021, divers in the PSW suction pit identified that the pump suction head was directly below the 1C PSW pump location standing vertically in its original position. The pump suction head was removed from the suction pit on September 3, 2021. Subsequent inspections after FM retrieval did not find any bolts/nuts or other material near the other pumps. Based on this, the potential for common cause failure due to foreign material (FM) (i.e., damage of all pumps due to FM intrusion) was investigated and resolved as not applicable. The other three pumps continue to run and provide rated flow.

The initial pump failure was a failure to run. The 1C pump had been running since 8/19/21 at 1616, about 7 days. The second pump failure was a fail to start following the pump maintenance. The choice of whether to adjust the Fail to Start or Fail to Run CCF events was based on events with the highest delta risk impact. The Fail to Run common cause is also modified indirectly by increasing the Qt base failure rate as discussed below. To provide an evaluation that is bounding, the PSW 1C evaluation that follows adjusts the Fail to Run CCF basic events and adjusts the PSW pump fail-to-run base reliability. A sensitivity case was also performed that does not adjust these values, as an estimate of a more realistic evaluation. A second sensitivity was conducted to determine the impact of applying the common cause to only the Fail to Start to verify potential compensatory actions and ensure the Failure to Run common cause is the bounding case. The third sensitivity conducted was based on an increased LOSP initiating event based on the potential of severe weather.

2.1.2 Quantification Truncation

To generate both the base and CT case risk, each hazard was quantified at the truncation levels below to ensure that the basic events for the 1C PSW pump were present.

Internal Events CDF – 1E-12
Internal Events LERF – 1E-13
Internal Flooding CDF – 1E-12
Internal Flooding LERF – 1E-13
Internal Fire CDF – 1E-12
Internal Fire LERF – 1E-12
Seismic CDF – 1E-11
Seismic LERF – 1E-11

2.1.3 Calculation Approach

Evaluation Assumptions

A) No credit is taken for the protection of equipment per site procedure NMP-OS-010. Since routine work on a number of components is prohibited per this process, the maintenance events for these could be set to zero in the assessment. This was not done and thus the results are conservative.

B) Existing components already out of service are not included in the base risk calculations. Additionally, increased failure to run probabilities for the running pumps (PSW 1A, 1B, 1D) are not included in the base risk. This results in a lower base risk and therefore a higher delta risk; thus, the risk calculations are conservative.

Risk Assessment Details

The base model used for this assessment is the Revision 8 Phoenix one-top model of record documented in calculation H-RIE-PHOENIX-U01, Revision 3. This model has the required quantification and support files set up to calculate either zero-maintenance or average maintenance risks. It contains flag events that allow NFPA-805 credited modifications to be turned on or off. This risk assessment takes no credit for NFPA-805 modifications yet to be installed.

The Revision 8 Phoenix Risk Model of record (PRM) contains internal events, internal flooding, internal fire, and seismic hazards. All other hazards have been screened out as being very low risk. The model can be evaluated one hazard at a time or with all hazards activated.

PRAQuant 5.2 was used for quantification. FTREX 2.0.0.1 64-bit wrapper was used as the quantification engine. The cutsets were calculated without recoveries applied and then, due to the complexity of the recovery rules and the limitations of the 32-bit version of QRecover provided with PRAQuant., QRecover version 10 was used externally to apply recovery rules.

The base risk used for this evaluation is taken directly from the PRM Rev 8 calculation. Existing out of service components were not included in the base case quantification. The components currently out of service in addition to PSW pump 1C were included only in the pump failed cases. This adds conservatism to the evaluation since the base case is lower and the delta risk is higher than if the out of service components were included in the base case.

The proposed technical specification change involves unavailability of the PSW Pump 1C. The revised CDF and LERF values for the CT configurations are obtained by re-quantifying the base PRA model with the identified events set, as shown below, in a flag file.

The evaluation of ICCDP and ICLERP for this condition is determined as shown below:

The ICCDP associated with PSW Pump 1C OOS for a new CT is given by:
ICCDP1C= (CDF1C - CDFBASE) x CTNEW
where

CDF1C = the annual average CDF calculated with PSW Pump 1C OOS (and other currently OOS equipment)

CDFBASE = 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 was not included in the base case values.

CTNEW = 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 TS 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.

The guidance provided in Regulatory Guide 1.200, Revision 2, (Section C.4.2) requires the following items be addressed in documentation submitted to the NRC to demonstrate the technical adequacy of PRA models utilized for the application:

- Identification of permanent plant changes (such as design or operational practices) that have an impact on the PRA but have not been incorporated in the PRA.

- The parts of the PRA used to produce the results are performed consistently with the version of the PRA Standard endorsed by RG 1.200.

- A summary of the risk assessment methodology used to assess the risk of the application, including how the PRA model was modified to appropriately model the risk impact of the application.

- Identifications of key assumptions and approximations in the PRA relevant to the results used in the decision-making process.

- A discussion of the resolution of peer review or self-assessment findings and observations that are applicable to the parts of the PRA required for the application.

- Identification of parts of the PRA used in the analysis that were assessed to have capability categories less than that required for the application.

2.1.4 Common Cause Adjustments

The CCFA values were obtained from the RR file and checked against Hatch Rev 5 version 2 data update files, as these are the values used in the Rev 8 model. Both the updated fail to run and fail to start updated CCFA values were calculated to include in both the bounding and sensitivity case. Using the methodology in RG 1.177,

- For PSW pump failure to start (FTS), events of the form CC-SW-*, Qt is 1.79E-06 per hour and MT is 1 demand.
  - CCFA2 (2/4) = 1.01E-02. The basic events for 2/4 are increased from 1.50E-05 to 1.01E-02
  - CCFA3 (3/4) = 2.16E-03. The basic events for 3/4 are increased from 3.20E-06 to

2.16E-03
- CCFA4 (4/4) = 3.64E-03. The basic event for 4/4 is increased from 5.39E-06 to 3.64E-03

- For PSW pump failure to run (FTR), events of the form CC-PS-*, Qt is 1.48E-03 and MT is 24 hours.
  - CCFA2 (2/4) = 2.70E-03. The basic events for 2/4 are increased from 1.16E-07 to 2.70E-03
  - CCFA3 (3/4) = 4.45E-04. The basic events for 3/4 are increased from 1.91E-08 to 4.45E-04
  - CCFA4 (4/4) = 1.37E-03. The basic event for 4/4 is increased from 5.89E-08 to 1.37E-03

The events below contain the 1C pump in the CCF group and are inserted into a flag file as shown for the respective scenario:

FTS Sensitivity Case:
CC-SW-6 PROB 1.01E-02
CC-SW-8 PROB 1.01E-02
CC-SW-10 PROB 1.01E-02
CC-SW-11 PROB 2.16E-03
CC-SW-13 PROB 2.16E-03
CC-SW-14 PROB 2.16E-03
CC-SW-15 PROB 3.64E-03

FTR Bounding Case:
CC-PS-5 PROB 2.70E-03
CC-PS-6 PROB 2.70E-03
CC-PS-7 PROB 2.70E-03
CC-PS-11 PROB 4.45E-04
CC-PS-12 PROB 4.45E-04
CC-PS-13 PROB 4.45E-04
CC-PS-15 PROB 1.37E-03

Since none of these events are in the recovery files, flag files were used to adjust the values and the hazard models re-quantified. Because a pump cannot fail to start and fail to run, only one of the above groups is adjusted in the flag file. The CCF events in the other group related to the failed component should be set to False.

Impact of potentially degraded performance of the in-service PSW pumps

The remaining three PSW pumps have been running continuously since PSW 1C has been out of service. To account for uncertainty associated with failure probabilities of the running pumps, the value of the PSW pump Fails to Run type code P1 OR was increased by a factor of three. Although this is independent of the PSW 1C failure, the adjustment was not included in the base risk case, and thus the Delta risk results are conservative. Because the model uses calculations for the basic events in a common cause group, changing the Qt value propagates to all related basic events. The nominal mean value of P1 OR is 1.79E-06/hr, with a gamma distribution. The using the information used in the Bayesian update for the P1 OR type code, the calculated alpha parameter is 2.292 and the beta parameter is 1282045. Using the Excel function GAMMA.INV, the 95% value is 4.064E-06/hr or a factor of 2.27 times the mean value. For this evaluation, a copy of the fault tree, RR file and recovery fault tree were created and P1 OR was increased by a factor of 3, to 5.3634E-06/hr. The final PSW 1C Case Runs used these files and the CCF adjustments above.

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
TIER I ASSESSMENT AND RESULTS FOR UNIT 1

Hazard	Base risk (/yr)	New risk (/yr)	Delta Risk (/yr)	Integrated Value for 15-day extension (45-day total CT)
Internal Events CDF	4.53E-06	1.87E-05	1.42E-05	
Internal Flooding CDF	3.73E-07	8.32E-07	4.59E-07	
Internal Fire CDF	5.86E-05	6.20E-05	3.42E-06	
Seismic CDF	9.21E-07	8.98E-07	0.00E+00	
Total Individual CDF	6.44E-05	8.25E-05	1.80E-05	
OTMHM CDF	6.44E-05	8.25E-05	1.80E-05	2.22E-06 (ICCDP)
Internal Events LERF	3.19E-07	6.73E-07	3.53E-07	
Internal Flooding LERF	6.12E-09	1.53E-08	9.18E-08	
Internal Fire LERF	3.78E-06	3.82E-06	4.00E-08	
Seismic LERF	2.77E-07	2.50E-07	0.00E+00	
Total Individual LERF	4.38E-06	4.75E-06	3.76E-07	

OTMHM LERF	4.38E-06	4.96E-06	3.76E-07	4.63-08 (ICLERP)

Note: There was a slight decrease in seismic risk, which was not credited. The seismic decrease impact on overall plant risk was minor when compared to overall increase in other hazards.

Assessment of off-line modes

Hatch does not have a quantitative process for evaluating low power and shutdown risk. 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. PSW provides motor cooling for RHR Service Water pumps during shutdown. Since RHRSW is used for shutdown cooling, and RHRSW is cooled by PSW, there is some risk involved with going into lower modes; however, that is not quantified or discussed any further in this assessment. The risk increase during shutdown is thus not zero and this is another reason for performing the repair on-line.

Compensatory Measures Discussion

Risk Insights

Risk insights from this configuration were examined by comparing the change in Fussell-Vesely (FV) values between the base and configuration specific importance rankings. The importance rankings were generated from the global all-hazards cutsets using CAFTA. A spreadsheet called Importances.xlsx contains the base model importance reports and the configuration specific importance reports. FV values that increased by a factor of three(200%) or more were examined to see what basic events contributed more to overall risk due to the failed pump. Because components and operator actions may be represented by multiple basic events, the overall risk increase is the sum of the individual event F-V terms. In the Importances.xlsx spreadsheet, there are 609 basic events that increased by 200% or more. The events and components that become more important are associated with the other PSW pumps, HPCI, RCIC, the 1B diesel and the Containment Hardened Vent. In addition, the %IE-LOSP initiating event became more important. These are discussed in more detail below.

Tier 2 Evaluation – Avoidance of risk significant configurations.

RG 1.177 requires an examination of other components that, in combination with the component already out of service, could result in a risk significant configuration. Maintenance Rule risk ranks components primarily on RAW, so that was chosen for component measures. Time Critical Operator Action guidance from the PWROG ranks operator actions using Birnbaum, so that risk measure is also chosen. The Importances.xlsx spreadsheet was used to identify basic events where the selected importance measure increased by a factor of three (delta risk greater than 200%) over the base case measure. These basic events and interpretation from the model are then used to help determine the Compensatory Measures.

For the PSW 1C pump, the items below had Fussell-Vesely risk increases greater than a factor of three.

Components:

Component	Description
1E41C001	HPCI Pump
1E51C001	RCIC Pump
1R43S001B	1B Diesel Generator
1T48F082	Torus Vent SGTS Isolation Valve
1T48F081	Torus Vent SGTS Isolation Valve

HPCI and RCIC can run without PSW to the room coolers for a substantial period of time.

Initiating Events:

Initiating Component	Description
1S11S004	Start Up Auxiliary Transformer 1C (230kV/4160)
1S11S005	Start Up Auxiliary Transformer 1D (230kV/4160)
1S11S052	Start Up Auxiliary Transformer 1E (230kV/4160)

All three (3) operable PSW pumps are currently running and would only shutdown and restart for Loss of Offsite Power events. The three Unit 1 startup transformers are added to the list above to address the %IE-LOSP initiator.

Operator Actions:

Operator Action ID	Description
OPHERHRSWPMPCL (and -F, -L, -S versions)	Restore/Crosstie RHRSW pump motor cooling (34AB-P41-001-1)
OPHEPSWXTIE (and -F, -L , -S versions)	Crosstie Reactor Building Division 2 header to Division 1 header.

The above operator actions are associated with loss of service water to the RHRSW pumps and to the containment coolers, based on one of two pumps in division 1 already out of service.

<u>Compensatory Measures</u>

Because the ICCDP is slightly above 1E-06 , compensatory measures are required. These measures are based on procedural protection, operation of redundant functions, and recommended actions based on the risk insights from the risk significant configurations above.

The following equipment is protected as required by SNC Procedure NMP OS 010-002 (Reference 1) for 1C PSW pump out-of-service:

- o 1A PSW Pump
- o 1E 4160V Frame 3 (power supply to 1A PSW Pump)
- o 1A PSW Pump Control Switch

In addition, travelling water screen 1W32E003B will be placed in RUN to compensate for the 1A screen (1W33E003A) being out of service if it is removed from service.

Based on the operator actions and equipment above, the following compensatory actions are recommended prior to exceeding the original 30-day LCO.

- The operator actions are contained in procedure 34AB-P41-001-1 and the recommended compensatory action is to have an Operations Standing Order for shift briefing on the procedure associated with these actions.
  - Section 4.1.7 Crosstie Cooling Sources. This ensures that adequate cooling will remain available for the RHRSW pump
  - Section 4.1.9 Operating Strategy For Event Mitigation. This is preparation for potential loss of an additional PSW pump
- Protect HPCI and limit work to TS required surveillances only. Note that HPCI was out of service during the initial 30 day completion time, and has since been returned to service.
- Protect RCIC and limit work to TS required surveillances only.
- Do not perform maintenance on 1T48F081 or 1T48F082, the Torus Ventilation SGTS Isolation valves used as the Containment Hardened Vent path.
- Protect the 1B diesel generator and the Standby Service Water (SSW) pump 2P41C002 that provides normal cooling to the 1B DG and limit work to TS required surveillances only.
- Protect the three Unit 1 startup transformers and their associated 230KV breakers.
- Verify that the portable FLEX pumps that can be used to provide water to the PSW header in the reactor building are available.

<u>Tier 3 Evaluation – A(4) Maintenance Rule configuration risk management impact.</u>

Pump PSW 1C was input into the on-line configuration risk management (CRM) program. The Hatch CRM 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 PSW 1C pump failure were left out of service for this evaluation to ensure the calculation is conservative. With the 1C pump out of service, the increase in risk is minimal as shown on Attachment 1. Because 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 and the a(4) process of assessing and managing that risk will adequately control the evolution. This includes any risk management actions that may occur during risk evaluations for rigging and lifting the motor and pump near other components that are remaining in service.

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 3.7.2.A extension risk assessment.

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. 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 was performed using the criteria in ASME PRA Standard RA-Sa-2009 and concluded that all other external hazards (i.e., all external hazards other than internal fires and seismic) can be screened from applicability at Hatch. Therefore, there is no significant other external hazards risk contribution for this application.

2.3.2 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. 6]. 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 results indicate a one-time extension up to 45 days would not exceed the ICCDP and ICLERP risk limits. Additional compensatory measures would potentially reduce risk further, such as protected equipment processes and other identified activities that impact potential to reduce LOOP probability. 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 3.7.2.A CT extension assessment. Assumptions and Uncertainties for the Base Hazard Calculations

The purpose of this section is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for Condition 3.7.2.A CT extension assessment. The baseline internal events PRA, internal flooding PRA, fire PRA (FPRA) and seismic PRA 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 any items which may be directly relevant to Condition 3.7.2.A CT extension assessment, discuss the results, and to provide dispositions. No key assumptions or sources of uncertainty were identified that uniquely impact this application.

Several additional areas of uncertainty were also investigated.

1) The site has discovered several circuit breakers that are not well coordinated with downstream loads. The listing of breaker and loads was reviewed, and none of them impact the PSW pumps or the components that become more important with a pump out of service. The Hatch Fire PRA was built with the assumption that all credited power supplies in the Fire PRA were coordinated with their upstream breakers. After further review, it was identified that there is a subset of breakers that are in fact not coordinated with the upstream breakers. Memo, 05254-MEM-01, documents the initial review of the impact of these uncoordinated breakers. This memo evaluated these breakers in a conservative manner in that each breaker was failed in every scenario.

   The results presented in this memo include cutsets for both CDF and LERF for both units. For the purposes of the PSW 1C evaluation, the unit 1 cutsets are the ones of concern.

   To review the potential impact of the breaker coordination on the PSW 1C evaluation, the following steps were taken.

   - Use the ‘Delterm’ feature in CAFTA to identify the new cutsets introduced in the breaker coordination case
   - Start with the breaker coordination cutset and perform the Delterm function with the base cutset
   - Compress out the deltermed cutsets such that the only remaining cutsets are those that are introduced from the breaker coordination failures
   - Search the deltermed cutsets for each of the events set to logically TRUE in the PSW 1C evaluation
   - Identified in the “Model Quantification Details and Results” section.
   - The results of this review showed that the deltermed cutsets (both CDF and LERF) do not include any of the basic events in the PSW 1C evaluation. This supports that the uncoordinated breakers will have minimal risk impact on the PSW 1C quantification.

2) This LCO is being extended during the hurricane season. Loss of Offsite Power is one of the initiating events that becomes more important with a PSW pump out of service.

This is addressed by a sensitivity run that increases the weather portion of the LOSP initiator to its 95% value. This value was calculated previously for the Unit 1 Diesel Generator liner replacement LCO extension in SNC calculation PRA-BC-H-20-001.

3) The treatment of open phase protection (OPP) of the startup transformers is a potential source of uncertainty. Hatch did not add OPP equipment or initiating events to the PRA models, based on the event screening done in accordance with NEI guidance. Because of the arrangement of offsite power sources, the potential for OPP is very small. This does not represent a significant source of uncertainty for evaluating PSW pumps. This evaluation is contained in the Hatch OPP evaluation calculation PRA-BC-H-19-004.

Sensitivities

The failure of the PSW 1C pump after maintenance activities is not suspected to impact the other three PSW pumps, which remained running throughout the evolution. The runs without CCF adjustment are presented below.

Table 1 PSW 1C, No Adjusted CCF

Hazard	Base risk (/yr)	New risk (/yr)	Delta Risk (/yr)	Integrated Value for 15-day extension (45-day total CT)
Internal Events CDF	4.53E-06	4.98E-06	4.54E-07	
Internal Flooding CDF	3.73E-07	3.95E-07	2.18E-08	
Internal Fire CDF	5.86E-05	5.91E-05	5.15E-07	
Seismic CDF	9.21E-07	9.08E-07	0.00E+00	
Total Individual CDF	6.44E-05	6.54E-05	9.78E-07	
OTMHM CDF	6.44E-05	6.54E-05	1.00E-06	1.23E-07 (ICCDP)
Internal Events LERF	3.19E-07	3.42E-07	2.26E-08	
Internal Flooding LERF	6.12E-09	8.90E-09	2.78E-09	
Internal Fire LERF	3.78E-06	3.78E-06	6.70E-09	
Seismic LERF	2.77E-07	2.51E-07	0.00E+00	
Total Individual LERF	4.38E-06	4.38E-06	6.31E-09	

OTMHM LERF	4.38E-06	4.39E-06	1.10E-08	1.36E-09 (ICLERP)

CCF FTS instead of FTR and increased FTR Unreliability

Table 2 PSW 1C, FTS CCF and FTR Increased Unreliability

Hazard	Base risk (/yr)	New risk (/yr)	Delta Risk (/yr)	Integrated Value for 15-day extension (45-day total CT)
Internal Events CDF	4.53E-06	1.80E-05	1.34E-05	
Internal Flooding CDF	3.73E-07	6.38E-07	2.65E-07	
Internal Fire CDF	5.86E-05	5.93E-05	6.92E-07	
Seismic CDF	9.21E-07	9.25E-07	4.05E-09	
Total Individual CDF	6.44E-05	7.88E-05	1.44E-05	
OTMHM CDF	6.44E-05	7.89E-05	1.44E-05	1.78E-06 (ICCDP)
Internal Events LERF	3.19E-07	8.00E-07	4.81E-07	
Internal Flooding LERF	6.12E-09	1.63E-08	1.02E-08	
Internal Fire LERF	3.78E-06	3.81E-06	3.12E-08	
Seismic LERF	2.77E-07	2.51E-07	0.00E+00	
Total Individual LERF	4.38E-06	4.87E-06	4.96E-07	
OTMHM LERF	4.38E-06	4.87E-06	4.96E-07	6.12E-08 (ICLERP)

Increased weather impacts.

Calculation PRA-BC-H-20-001 performed for the Unit 1 Diesel Generator LAR extensions calculated an increase in the %IE-LOSP initiating event frequency using the 95% value for the weather portion. This increases the initiator frequency from 2.115E-02/yr to 3.378E-02/yr. The existing CDF FTR CCF cutset for Internal Events was opened and the frequency changed. This changed the CDF risk from 1.87E-05 to 2.20E-05. This is reflected below. The LERF risk increased from 6.73E-07 to 7.80E-07.

Table 3 PSW 1C, FTR CCF and FTR Increased Unreliability Increased LOSP Initiating Event Frequency

Hazard	Base risk (/yr)	New risk (/yr)	Delta Risk (/yr)	Integrated Value for 15-day extension (45-day total CT)
Internal Events CDF	4.53E-06	2.20E-05	1.75E-05	
Internal Flooding CDF	3.73E-07	8.32E-07	4.59E-07	
Internal Fire CDF	5.86E-05	6.20E-05	3.42E-06	
Seismic CDF	9.21E-07	2.50E-07	0.00E+00	
Total Individual CDF	6.44E-05	4.75E-05	2.13E-05	
				2.63E-06 (ICCDP)
Internal Events LERF	3.19E-07	7.80E-07	4.61E-07	
Internal Flooding LERF	6.12E-09	1.53E-08	9.18E-09	
Internal Fire LERF	3.78E-06	3.82E-06	4.00E-08	
Seismic LERF	2.77E-07	2.50E-07	0.00E+00	
Total Individual LERF	4.38E-06	4.75E-07	4.83E-07	
				5.95E-08 (ICLERP)

These risk increases are small and still within guidance.

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 3.7.2.A. The One-Top Multihazard Model (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 1 TS Condition 3.7.2.A for a one-time increase of the CT from 30 days to 45 days when the 1C PSW Pump 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. Additional sensitivity analysis show that the RG 1.177 thresholds are not challenged.

4.3 CONCLUSIONS

This analysis demonstrates the acceptability, from a risk perspective, of a change to the Hatch TS Condition 3.7.2A to increase the CT from 30 days to 45 days when the 1C PSW Pump is 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 2, 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] RBA-21-007-H – “PSW 1C Emergent Technical Specification Change”