Southern Nuclear

Regulatory Affairs

3535 Colonnade Parkway
Birmingham, AL 35243
205 992 5000 tel
205 992 7795 fax

November 29, 2019

Docket No.: 50-321 NL-19-1455

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 3.7.1
Regarding One-Time Extension of Completion Time for
RHRSW 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 3.7.1, “Residual Heat Removal Service Water (RHRSW) System,” Condition A, “One RHRSW 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 December 18, 2019 at 0845 eastern standard time (EST).

The one-time only change allows for continued repair and testing activities on the RHRSW pumps. The expiration date for the proposed allowance is based on the current 30-day Completion Time expiration at 0845 EST, December 3, 2019, 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 RHRSW System, pursuant to 10 CFR 50.91(a)(5). The Unit 2 RHRSW System is not affected by this proposed amendment.

SNC requests approval of the proposed license amendment as soon as possible and no later than December 2, 2019 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 pages, respectively, reflecting the proposed changes. Attachment 3 contains a markup of the TS Bases, for information only. Attachment 4 contains an evaluation of the risk impact and a discussion of the compensatory measures related to the changes in this amendment request.

In accordance with the SNC administrative procedures and the HNP quality assurance program manual, this proposed license amendment has been 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 Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29 day of November 2019.

Respectfully submitted,

Jamie M. Coleman

Jamie M. Coleman
Licensing Manager
Southern Nuclear Operating Company

JMC/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 Pages
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 3.7.1
Regarding One-Time Extension of Completion Time for
RHRSW 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) 3.7.1, “Residual Heat Removal Service Water (RHRSW) System,” Condition A, “One RHRSW 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 2.2 of Attachment 4 and as affirmed in the NRC’s safety evaluation are implemented, and would expire on December 18, 2019, at 0845 EST.

On November 3, 2019, at 0845 EST, HNP Unit 1 RHRSW pump 1B was declared inoperable and the pump was subsequently replaced. On November 21, 2019, the new RHRSW pump 1B was run for a pre-service test. During the test a loss of flow and pressure was observed, and the pump was shut down, thus failing its pre-service test and remaining inoperable. Divers performed inspections of the RHRSW pump 1B suction in the intake bay and reported no visual obstructions and no obvious issues with structural integrity.

Southern Nuclear Operating Company (SNC) then again replaced the pump with another new pump and performed extensive additional work, including rework of the discharge head. On November 28, 2019, the second new pump ran for approximately one minute before being shut down due to signs of overheating. This new pump issue is currently under investigation. The vendor is on site assisting with the investigation. At this time, SNC does not believe the failures of the two new pumps originate from a common cause.

The comprehensive repair work has been time-consuming; however, SNC has demonstrated due diligence by twice replacing the inoperable pump. SNC now expects that repair and testing will extend past the 30-day Completion Time of TS 3.7.1, Condition A and therefore requests additional time to make careful, prudent repairs with appropriate compensatory measures in place to return the HNP Unit 1 Division II RHRSW system to operable status. To provide allowance for additional unforeseen discoveries, such as the need to order new parts or to contract with a specialty vendor, SNC is requesting the Completion Time be temporarily extended from 30 days to 45 days.

2. Detailed Description

2.1 Emergency Circumstances

The emergency circumstances resulted from the failure or imminent failure of two new 1B RHRSW pumps during pre-service testing. The required Completion Time of TS 3.7.1, Condition A of thirty days is currently applicable and will expire on December 3, 2019. SNC can neither replace nor finish repairs of pump 1B by that time. Neither a routine nor an exigent amendment can be processed prior to December 3, 2019 at 0845 EST.

SNC requests an expedited review of the proposed license amendment in accordance with the provisions of 10 CFR 50.91(a)(5) based on the failure or imminent failure of the new RHRSW pumps during pre-service testing.

SNC has performed due diligence by replacing the RHRSW pump 1B twice. If the proposed license amendment is not approved, Unit 1 will be required to enter Mode 3 on December 3, 2019.

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

2.2 System Design and Operation

The RHRSW System is designed to provide cooling water for the Residual Heat Removal (RHR) System heat exchangers, required for a safe reactor shutdown following a design basis accident (DBA) or transient. The RHRSW System is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR System. The system is initiated manually from the control room.

The RHRSW System consists of two independent and redundant subsystems. Each subsystem is made up of a header, two 4,000 gpm pumps, a suction source, valves, piping, heat exchanger, and associated instrumentation. Either of the two subsystems can provide the required cooling capacity with two pumps operating to maintain safe shutdown conditions. The two subsystems are separated from each other by normally closed motor operated cross-tie valves, so that failure of one subsystem will not affect the operability of the other subsystem. Similarly, two manual isolation valves provide a second normally closed cross-tie between the two subsystems. The RHRSW System is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The RHRSW System is described in the HNP Unit 1 Updated Final Safety Analysis Report (UFSAR), Section 10.6. Both RHRSW subsystems cross-connect to the corresponding Plant Service Water (PSW) subsystem. The cross-connections between PSW and RHRSW are provided with manual, double isolation valves which are maintained closed except for periodic plant maintenance activities such as dead-leg flushing. The cross-connections are not used during normal or design basis accident conditions. The RHRSW-PSW cross-connections are only to be used in response to a beyond design basis external event.

Cooling water is pumped by the RHRSW pumps from the Altamaha River through the tube side of the RHR heat exchangers, and discharges to the circulating water flume. A minimum flow line from the pump discharge to the intake structure prevents the pump from overheating when pumping against a closed discharge valve.

The RHRSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. Analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown and include the evaluation of the long-term primary containment response after a design basis LOCA.

The safety analyses for long term cooling were performed for various combinations of RHR System failures. The worst-case single failure that would affect the performance of the RHRSW System is any failure that would disable one subsystem of the RHRSW System. As discussed in the HNP Unit 2 UFSAR, Section 15.4.10.1.1 for these analyses, manual initiation of the OPERABLE RHRSW subsystem and the associated RHR System is assumed to occur 10 minutes after a DBA. The RHRSW flow required to support the assumed heat removal rate is 3,750 gpm per pump with two pumps operating in one loop (subsystem) with up to 5% tubes plugged in the RHR heat exchanger. In such a case, the maximum suppression chamber water temperature and pressure are approximately 211.6°F and 27.8 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig.

An RHRSW subsystem is considered operable when two pumps are operable, and an operable flow path is capable of taking suction from the Intake Structure and transferring water to the RHR heat exchangers at the assumed flow rate. Additionally, the RHRSW crosstie valves (which allow the two RHRSW subsystems to be cross-connected) must be closed so that the failure of one subsystem will not affect the operability of the other subsystem.

2.3 Current Technical Specification Requirements

Currently, TS 3.7.1, Condition A, requires restoration of an inoperable RHRSW pump to operable status within 30 days.

2.4 Reason for the Proposed Change

Despite its diligent and prudent efforts, SNC has been unable to return the RHRSW pump 1B to operable service and now expects it will be unable to do so by expiration of the Completion Time for TS 3.7.1, Condition A (i.e., 0845 am EST, December 3, 2019) and would then be required to place HNP Unit 1 in Mode 3. 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 3.7.1 (added text in *italics*):

- Note added to Completion Time of Condition A:

  "*NOTE: A Completion Time of 45 days is permitted while the compensatory measures described in Attachment 4 of SNC letter NL-19-1455 dated November 29, 2019 are implemented. This allowance expires at 0845 EST on December 18, 2019.*"

- This proposed change would cause Condition D to move to the next page of the TS. SNC is proposing no wording changes to Condition D.

The one-time only change allows for continued repair and testing activities in a prudent fashion. The expiration date of December 18, 2019 at 0845 EST is based on the current 30-day Completion Time expiration at 0845 EST, December 3, 2019, plus the requested additional 15 days (45 days total). The allowance, which allows SNC to safely and

prudently address any additional unforeseen discoveries (such as the need to order new parts or to contract with a specialty vendor) would only apply as long as the compensatory measures described in Section 2.2 of Attachment 4 of this application are implemented.

3. Technical Evaluation

3.1 Defense-in-Depth

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

In addition, compensatory measures, as described in Section 2.2 of Attachment 4, 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 RHRSW System.

The RHR System and the UHS have sufficient capacity to function for design basis accidents. Assuming no additional failures, the UFSAR acceptance criteria for the design events will be met should such an event occur during the time that the RHRSW pumps are out of service.

3.3 Compensatory Measures

Attachment 4 of this license amendment request presents the evaluation of risk impacts and compensatory measures for the proposed amendment. The compensatory measures will ensure that RHRSW remains available during a design basis accident or plant transient.

3.4 Maintenance Rule Control

The RHRSW pumps are included under the HNP Maintenance Rule Program and function to transfer cooling water from the Altamaha River through the tube side of the RHR heat exchangers. Since the pumps' function is to operate during design basis accidents (DBAs) and plant transients, the pumps are on standby the majority of the time; thus, the RHRSW System is exempt from the Maintenance Rule unavailability hours.

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

As part of compliance with 10 CFR 50.65, performance is monitored against licensee-established goals. If the performance of the RHRSW 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.

3.5 Evaluation of Risk Impacts

The risks associated with a one-time extension of the HNP Unit 1 TS 3.7.1, "Residual Heat Removal Service Water (RHRSW) System," Condition A, "One RHRSW pump inoperable," 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 1).

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 2), and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications," Revision 1 (Reference 3).

Attachment 4 of this license amendment request presents the evaluation of risk impacts due to the proposed amendment. The evaluation in Attachment 4 for the 1B RHRSW pump bounds either of the Division II RHRSW pumps (1B or 1D).

4. Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

The design of the RHRSW 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 RHRSW System is described in the HNP Unit 1 UFSAR Section 10.6.

The proposed amendment does not delete requirements associated with the RHRSW System and limiting condition for operation (LCO) 3.7.1 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).

HNP Unit 1 was not licensed to the 10 CFR 50, Appendix A, General Design Criteria (GDC). HNP Unit 1 was licensed to the applicable Atomic Energy Commission (AEC) preliminary general design criteria identified in Federal Register 32 FR 10213, published July 11, 1967 (ADAMS Accession No. ML043310029). The AEC proposed criteria applicable to the HNP Unit 1 RHRSW System were compared to the 10 CFR 50, Appendix A, GDC, as documented in the Hatch Updated Final Safety Analysis Report (UFSAR), Appendix F, "Conformance to the Atomic Energy Commission (AEC) Criteria," as discussed below.

The applicable GDC are Criterion 34, “Residual Heat Removal,” and Criterion 44, “Cooling Water”.

Criterion 34 states:

> “A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.”

Criterion 44 states:

> “A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.”

Design evaluations of GDC 34 and of GDC 44 are included in Section F.3 of the HNP Unit 1 UFSAR, "Evaluation with Respect to 1971 General Design Criteria."

Following implementation of the proposed change, HNP Unit 1 will remain in compliance with applicable AEC design criteria as described in the HNP Unit 1 UFSAR.

4.2 Precedent

The proposed change is similar to NRC-approved License Amendment 165 issued under exigent circumstances to Vogtle Electric Generating Plant (VEGP), Unit 2 on December 21, 2016 (NRC Agencywide Documents Access and Management System (ADAMS) Accession No. ML16354A133). This amendment added a note to extend the

Completion Time on a one-time basis for VEGP Unit 2 Condition D.2.2 of LCO 3.7.9 to 77 days from 31 days to allow for refurbishment of the 2A nuclear service cooling water (NSCW) transfer pump. The accompanying NRC safety evaluation concluded the license amendment involved no significant hazards consideration.

4.3 No Significant Hazards Consideration Analysis

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests an amendment to Hatch Nuclear Plant (HNP) Unit 1 renewed facility operating license DPR-57. The proposed amendment would revise Condition A, of LCO 3.7.1, of Technical Specification (TS) 3.7.1, “Residual Heat Removal Service Water (RHRSW) System,” to extend the Completion Time from 30 days to 45 days. This proposed allowance would expire on December 18, 2019 at 0845 EST and be effective only while the compensatory measures described in Section 2.2 of Attachment 4 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 amendment extends the Completion Time allowed to make repairs and test the Division II Residual Heat Removal Service Water (RHRSW) System with appropriate compensatory measures in place and available. 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 Residual Heat Removal (RHR) 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 repair and testing of a pump do not involve any unanalyzed modifications to the design or operational limits of the RHRSW or RHR Systems. The redundant RHRSW subsystem and compensatory measures allowed by the Technical Specifications will remain unaffected. Therefore, no new failure modes or

accident precursors are created due to pump repair during the extended Completion Time.

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 RHRSW and RHR Systems will remain capable of mitigating the consequences of a design basis event such as a LOCA.

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

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

4.4 Conclusions

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

5. Environmental Consideration

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

6. References

1. Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2, dated March 2009
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 2, dated April 2015
3. Regulatory Guide 1.177, *An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications*, Revision 1, dated May 2011

Edwin I. Hatch Nuclear Plant Unit 1

Emergency License Amendment Request for Technical Specification 3.7.1 Regarding One-Time Extension of Completion Time for RHRSW Pump Inoperable

Attachment 1

HNP Unit 1 Technical Specification Marked-up Page

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHRSW pump inoperable.	A.1 Restore RHRSW pump to OPERABLE status.	------------NOTE----------- A Completion Time of 45 days is permitted while the compensatory measures described in Attachment 4 of SNC letter NL-19-1455 dated November 29, 2019 are implemented. This allowance expires at 0845 EST on December 18, 2019. ------------------------------ 30 days
B. One RHRSW pump in each subsystem inoperable.	B.1 Restore one RHRSW pump to OPERABLE status.	7 days
C. One RHRSW subsystem inoperable for reasons other than Condition A.	-----------------NOTE----------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by RHRSW System. --- C.1 Restore RHRSW subsystem to OPERABLE status.	 7 days

(continued)

Edwin I. Hatch Nuclear Plant Unit 1

Emergency License Amendment Request for Technical Specification 3.7.1
Regarding One-Time Extension of Completion Time for
RHRSW Pump Inoperable

Attachment 2

HNP Unit 1 Revised Technical Specification Pages

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHRSW pump inoperable.	A.1 Restore RHRSW pump to OPERABLE status.	------------NOTE----------- A Completion Time of 45 days is permitted while the compensatory measures described in Attachment 4 of SNC letter NL-19-1455 dated November 29, 2019 are implemented. This allowance expires at 0845 EST on December 18, 2019. ------------------------------- 30 days
B. One RHRSW pump in each subsystem inoperable.	B.1 Restore one RHRSW pump to OPERABLE status.	7 days
C. One RHRSW subsystem inoperable for reasons other than Condition A.	------------------NOTE----------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by RHRSW System. --- C.1 Restore RHRSW subsystem to OPERABLE status.	 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
E. Both RHRSW subsystems inoperable for reasons other than Condition B.	-----------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by RHRSW System. -- E.1 Restore one RHRSW subsystem to OPERABLE status.	8 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program

Edwin I. Hatch Nuclear Plant Unit 1

Emergency License Amendment Request for Technical Specification 3.7.1 Regarding One-Time Extension of Completion Time for RHRSW Pump Inoperable

Attachment 3

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

BASES

ACTIONS ~~(continued)~~

A.1 (continued)

The Completion Time is modified by a Note indicating allowance to extend the Completion Time from 30 days to 45 days from November 3, 2019 until 0845 EST on December 18, 2019 while the compensatory measures described in Attachment 4 of SNC letter NL-19-1455 are implemented. This change was approved by the NRC in early December 2019 to allow prudent time to repair and test the pumps.

B.1

With one RHRSW pump inoperable in each subsystem, if no additional failures occur in the RHRSW System, and the two OPERABLE pumps are aligned by opening the normally closed cross tie valves (i.e., after an event requiring operation of the RHRSW System), then the remaining OPERABLE pumps and flow paths provide adequate heat removal capacity following a design basis LOCA. However, capability for this alignment is not assumed in long term containment response analysis and an additional single failure in the RHRSW System could reduce the system capacity below that assumed in the safety analysis. Therefore, continued operation is permitted only for a limited time. One inoperable pump is required to be restored to OPERABLE status within 7 days. The 7 day Completion Time for restoring one inoperable RHRSW pump to OPERABLE status is based on engineering judgment, considering the level of redundancy provided.

C.1

Required Action C.1 is intended to handle the inoperability of one RHRSW subsystem for reasons other than Condition A. The Completion Time of 7 days is allowed to restore the RHRSW subsystem to OPERABLE status. With the unit in this condition, the remaining OPERABLE RHRSW subsystem is adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE RHRSW subsystem could result in loss of RHRSW function. The Completion Time is based on the redundant RHRSW capabilities afforded by the OPERABLE subsystem and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in an inoperable RHR shutdown cooling subsystem. This is an exception to LCO 3.0.6

(continued)

Edwin I. Hatch Nuclear Plant Unit 1

Emergency License Amendment Request for Technical Specification 3.7.1
Regarding One-Time Extension of Completion Time for
RHRSW 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 3.7.1.A from 30 days to 92 days for Unit 1 in order to allow for repair of the 1B RHRSW Pump. This analysis bounds either of the Division II RHRSW pumps (1B or 1D). 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 1

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 permanent changes, usually associated with changing CT.

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

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

1. *The licensee has demonstrated that implementation of the one-time only TS CT change impact on plant risk is acceptable (Tier 1):*
   - *ICCDP of less than 1.0x10^{-6} and an ICLERP of less than 1.0x10^{-7}, or*
   - *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 Section 3.1 which directs the licensee to define the treatment of the scope of risk contributors (i.e., internal initiating events, external initiating events, and modes of power operation at the time of the initiator). Discussion of these risk contributors are as follows:

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

1.5 Hatch PRA MODELS

This section addresses the requirements of Section 3.1 of RG 1.200, Revision 2 [Ref. 1] which directs the licensee to identify the portions of the PRA used in the analysis.

The PRA analysis uses the Rev. 6 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. It also corrects several model errors found 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 would impact the overall CDF and LERF and would need to be addressed further. To confirm this, the PRA impact was also evaluated using the draft Revision 7 model that addresses specific issues identified in the Hatch NFPA-805 and 50.69 RAI responses.

The Revision 6 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 but 2 of the F&Os have been closed. Two F&Os associated with internal flooding documentation are still open, however they do not impact this evaluation since they are not technical issues with the model. A review of the quantification and uncertainty notebooks for each hazard model did not find any assumption or uncertainty that was not addressed by running the sensitivity case using both the Rev 6 and Rev 7 models. For this evaluation, an additional sensitivity run was performed that demonstrates the single-top model provides adequate information and that seismic contributions are minimal and can be ignored. At present, the seismic model cannot be quantified at a truncation level low enough to get adequate results from the other hazards, so the seismic model is not included in the single top evaluation. However, the seismic contribution was shown to be in the low E-08 range when quantified separately and thus does not significantly impact the results.

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

Table 2.1-1

HATCH CDF AND LERF BASE RISK METRICS

Hazard(s)	Risk (1/yr)
OTMHM CDF w/o Seismic	5.246E-05
OTMHM LERF w/o Seismic	4.072E-06

The general configuration for the extended CT is Hatch at-power on both units with the 1B RHRSW Pump out of service. 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, including through the use of the Configuration Risk Management Program. Section 2.2 discusses compensatory actions to support the plant configuration during the extended completion time.

The PRA model was quantified using the base “average test and maintenance” PRA model with the 1B RHRSW Pump maintenance and random failure events set to TRUE. The average test and maintenance model represents baseline assumed maintenance frequencies for all components with the exception of Technical Specification violations that are normally excluded in the disallowed maintenance (mutually exclusive) logic in the base PRA model. Due to the time frame of the extension request, no specific maintenance terms are restricted in the quantification. Equipment currently out of service (associated with 1C Station Service Air Compressor (SAC), RB Working Floor Area Cooler, and Traveling Water Screen) is included in the risk change calculations. This configuration is represented in the PRA by setting specific flags as shown in Table 2.1-2. Adjustments for common cause factors associated with the RHRSW 1B pump are also included.

Table 2.1-2

EXTENDED CT CONFIGURATION REPRESENTATION

BASIC EVENT / GATE	DESCRIPTION	VALUE
CC-HS-1	1/4, P8SS1E11C001B	T
CC-HS-19	1/4, P8SR1E11C001B	T
CC-HS-1-RSP	1/4, P8SS1E11C001B (RSP)	T
CC-HS-19-RSP	1/4, P8SR1E11C001B	T
MNUNHS_PUMPB	1B RHRSW Pump Maintenance	T
CC-IA-13	SAC 1C	T
CC-IA-23	SAC 1C	T
MNUNIA_CMPSC	SAC 1C	T
FNOR1T41B015	U1 RB Working Floor Area Cooler	T
MNUN1W33E003A	Upstream Traveling Water Screen	T
CC-HS-2	1/4, P8SS1E11C001D	5.9885E-04
CC-HS-3	1/4, P8SS1E11C001A	5.9885E-04
CC-HS-4	1/4, P8SS1E11C001C	5.9885E-04
CC-HS-5	2/4, P8SS1E11C001B P8SS1E11C001D	0.00
CC-HS-8	2/4, P8SS1E11C001D P8SS1E11C001A	5.163E-03
CC-HS-9	2/4, P8SS1E11C001D P8SS1E11C001C	5.163E-03
CC-HS-6	2/4, P8SS2E11C001B P8SS2E11C001A	0.00

BASIC EVENT / GATE	DESCRIPTION	VALUE
CC-HS-7	2/4, P8SS1E11C001B P8SS1E11C001C	0.00
CC-HS-10	2/4, P8SS1E11C001A P8SS1E11C001C	5.163E-03
CC-HS-11	3/4, P8SS1E11C001B P8SS1E11C001D P8SS1E11C001A	0.00
CC-HS-12	3/4, P8SS1E11C001B P8SS1E11C001D P8SS1E11C001C	0.00
CC-HS-14	3/4, P8SS2E11C001D P8SS2E11C001A P8SS2E11C001C	7.3057E-04
CC-HS-13	3/4, P8SS1E11C001B P8SS1E11C001A P8SS1E11C001C	0.00
CC-HS-15	4/4, P8SS1E11C001B P8SS1E11C001D P8SS1E11C001A P8SS1E11C001C	0.00
CC-HS-16	1/4, P8SR1E11C001C	6.6661E-04
CC-HS-17	1/4, P8SR1E11C001A	6.6661E-04
CC-HS-18	1/4, P8SR1E11C001D	6.6661E-04
CC-HS-20	2/4, P8SR1E11C001C P8SR1E11C001A	1.33E-03
CC-HS-22	2/4, P8SR1E11C001C P8SR1E11C001B	0.00
CC-HS-21	2/4, P8SR1E11C001C P8SR1E11C001D	1.33E-03
CC-HS-24	2/4, P8SR1E11C001A P8SR1E11C001B	0.00
CC-HS-23	2/4, P8SR1E11C001A P8R1E11C001D	1.33E-03
CC-HS-25	2/4, P8SR1E11C001D P8SR1E11C001B	0.00
CC-HS-28	3/4, P8SR1E11C001C P8SR1E11C001D P8SR1E11C001B	0.00
CC-HS-27	3/4, P8SR1E11C001C P8SR1E11C001A P8SR1E11C001B	0.00
CC-HS-26	3/4, P8SR1E11C001C P8SR1E11C001A P8SR1E11C001D	1.6756E-04
CC-HS-29	3/4, P8SR1E11C001A P8SR1E11C001D P8SR1E11C001B	0.00
CC-HS-30	4/4, P8SR1E11C001C P8SR1E11C001A P8SR1E11C001D P8SR1E11C001B	0.00

2.1.2 Quantification Truncation

The OTMHM average maintenance model was quantified at truncations for 1E-11 CDF and 1E-12 for LERF. To ensure that the single-top model is suitable for this evaluation, a sensitivity

case with additional out of service equipment was run and the hazards were quantified individually, summed, and compared to the single top results. The results were satisfactory. Since the seismic values would not quantify at the chosen truncation levels, the individual hazard results (<1E-8 CDF) were used to exclude them from the OTMHM calculation.

2.1.3 Calculation Approach

The proposed technical specification change involves unavailability of the 1B RHRSW Pump. The revised CDF and LERF values for the CT configurations are obtained by re-quantifying the base PRA model with all of the identified events set as shown in Table 2.1-2.

The evaluation of ICCDP and ICLERP for this condition is determined as shown below: The ICCDP associated with RHRSW Pump 1B OOS for a new CT is given by:

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

where

CDF_{1B} = the annual average CDF calculated with RHRSW Pump 1B OOS and other currently OOS equipment assuming the configuration listed in Table 2.1-2 (all quantified hazards)

CDF_{BASE} = baseline annual average CDF with average unavailability for all equipment. This is the CDF result of the baseline PRA (all quantified hazards). Currently OOS equipment other than the RHRSW Pump 1B was not included in the base case values.

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

Note: ICCDP is a dimensionless probability.

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

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

2.2 OTMHM Quantification

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

Table 2.2-1

OTMHM RISK ASSESSMENT INPUT PARAMETERS AND RESULTS FOR UNIT 1

Input Parameter	Value
$CDF_{BASE\ wo\ seismic}$	5.246E-05
$CDF_{1B\ wo\ seismic}$	5.967E-05
Delta CDF	**7.21E-06**
$LERF_{BASE\ wo\ seismic}$	4.072E-06
$LERF_{1B\ wo\ seismic}$	4.72E-06
Delta LERF	**6.48E-07**

Compensatory Measures Discussion

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

RG 1.177 requires a Tier 2 examination of other components that, in combination with the component already out of service, could result in a risk significant configuration. Only two basic events had a RAW increase greater than 2.0; these were a common cause failure and a HEP combination. Therefore, there are no single components that have a significantly larger risk with the RHRSW 1B pump out of service.

Since the Unit 1 A and C RHRSW pumps are powered by the Unit 1 A and B diesels, entry into a 14-day RAS for any of the five diesels should avoided due to the requirement that the B diesel be locked to the unit with the diesel out of service. This is already an existing prohibition in the on-line configuration maintenance procedure NMP-GM-031.

RG 1.177 also requires a Tier 3 examination of the (a)(4) Maintenance Rule configuration risk impact. The 1B RHRSW pump was input into the on-line configuration risk management (CRM) program as a “what-if” evaluation. The Hatch CRM program calculates both the instantaneous and integrated risk and CRM risk levels are based on integrated risk levels. The

components already out of service prior to the RHR pump discovery were left out of service for this evaluation to ensure the calculation is conservative. With the 'B' pump out of service, the increase in risk is minimal. The CRM program uses the same hazard models that were used for this evaluation, and since the a(4) process evaluates planned work as well as current configurations, it will identify any potential high-risk conditions during the extended CT. The a(4) process of assessing and managing that risk will adequately control the evolution and risk management actions will be generated as necessary.

2.3 EXTERNAL EVENTS

2.3.1 Assessment of Relevant Hazard Groups

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

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

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

2.3.2 Seismic

During the truncation study, it was noted that the seismic hazard would not quantify at the truncation level used for the other hazards, and therefore an all hazard quantification using the OTMHM was not possible while maintaining fidelity of the other, more substantial hazards if seismic was included. However, the individual hazard results show that the seismic contribution was negligible for the proposed configuration. Therefore, seismic was excluded from the all-hazard OTMHM quantification and therefore the delta risk values.

2.3.4 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. 10]. 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.

2.5 RESULTS COMPARISON TO ACCEPTANCE GUIDELINES

The values from Table 2.2-1 show that the delta CDF is more limiting. The time to reach 1E-06 ICCDP would be approximately 50.6 days total. The time to reach 1E-05 ICCDP would be >1 yr. The time to reach the Unit 1 refueling outage, at 92 days from the start of the LCO entry, would result in approximately 1.82E-6 ICCDP. Using equation 2-2 with the limiting delta CDF, the ICCDP for any given number of days can be determined.

The results indicate a one-time extension up to the next Hatch Unit 1 refueling outage would not exceed the ICCDP and ICLERP risk limits. Additional compensatory measures would potentially reduce risk further, such as protected equipment and abstaining from entry into the diesel generator 14-day LCO or other activities that impact diesel generator availability. The additional compensatory measures are not accounted for in the quantification.

2.6 UNCERTAINTY ASSESSMENT

The purpose of this section is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for Condition 3.7.1.A CT extension assessment. The baseline internal events PRA, internal flooding PRA, and fire PRA (FPRA) models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to Condition 3.7.1.A CT extension assessment, discuss the results, and to provide dispositions. As discussed in Section 1.5, no key assumptions or sources of uncertainty were identified that uniquely impact this application. However, some outstanding plant changes that are not accounted for in Rev. 6 of the OTMHM could impact the application. A sensitivity was performed by running a given configuration in both Rev. 6 and Rev. 7 models. The configuration consisted of the 1B RHRSW pump outage, other currently out of service equipment, an additional hypothetical RHRSW pump out of service, and common cause factor adjustments. The results show an improvement (decrease) of approximately 1E-5 in the total OTMHM-quantified delta CDF in the Rev. 7 model. Therefore, it was deemed acceptable to use the published Rev. 6 model in the risk assessment for the proposed configuration without making modeling adjustments.

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 3.7.1.A. The OTMHM is comprised of various hazards PRA models that can be quantified simultaneously or individually. Each hazard (internal events, internal flooding, fire, and seismic) has been peer reviewed. The most up-to-date assessments of PRA technical adequacy (including peer review status, F&O closure status, scope, fidelity, capability, and maintenance/update practices) was provided to the NRC previously for the Hatch 50.69 LAR (ADAMS Accession Number ML18158A583) and subsequent RAI responses (ML19197A097); and also the NFPA-805 LAR (ML18096A955) and subsequent RAI responses

(ML19280C812). Additionally, those submittals contain the most up-to-date description of the other external hazards assessment. As of now, there are only two remaining open F&Os associated with internal flooding documentation. Since the open F&Os do not affect the quantification of the results for this application, no additional sensitivity assessments are required.

4.0 SUMMARY AND CONCLUSIONS

This analysis evaluates the acceptability, from a risk perspective, of a change to the Hatch Unit 1 TS Condition 3.7.1.A for a one-time increase of the CT from 30 days to 45 days when the RHRSW Pump 1B is inoperable.

The analysis examines a range of risk contributors including internal events, internal flooding, fire, seismic, shutdown risk and other external hazards. With the exception of seismic risk, the configuration was quantified using the OTMHM model and compared to the base risk to obtain delta CDF and LERF values. Seismic was not included after a sensitivity showed that the hazard was not sensitive to the proposed configuration and the contribution to the delta values was negligible.

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. This combined with

effective compensatory measures to maintain lower risk ensures that the TS change meets the intent of the ICCDP and ICLERP acceptance guidelines.

4.3 CONCLUSIONS

This analysis demonstrates the acceptability, from a risk perspective, of a change to the Hatch TS Condition 3.7.1.A to increase the CT from 30 days to 45 days when the RHRSW Pump 1B is unavailable. This analysis bounds either of the Division II RHRSW pumps (1B or 1D).

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.

The assessment of risk from OTMHM Rev. 6 identified potential compensatory measures that would help to reduce the overall risk during the proposed configuration.

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, March 2009.

[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 1, May 2011.

[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-IEIF-U00-011 – “Internal Events Uncertainty”

[7] H-RIE-IF-U00-006 – “Internal Flooding Uncertainty”

[8] H-RIE-FIREPRA-U00-015 – “Internal Fires Uncertainty”

[9] H-RIE-SEISMIC-U00-001-002 – “Seismic Model Uncertainty”

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

[11] RBA-H-19-003 – “RHRSW Pumps 1B and 1D Emergent Technical Specification Change”