From: Lamb, John
Sent: Wednesday, September 22, 2021 11:24 AM
To: Enfinger, Timothy Lee; Joyce, Ryan M.
Subject: REQUEST FOR ADDITIONAL INFORMATION - Hatch, Unit 1, Emergency TS 3.7.2 LAR

Importance: High

Tim and Ryan,

By letter dated September 20, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21264A003), the Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted a license amendment request (LAR) for the Edwin I. Hatch Nuclear Plant (Hatch), Unit 1. The proposed amendment would revise the Hatch, Unit 1 Technical Specification (TS) requirements of TS 3.7.2, “Plant Service Water (PSW) System and Ultimate Heat Sink (UHS).” Specifically, the proposed amendment would revise Limiting Condition for Operation (LCO) 3.7.2, Condition A, “One PSW pump inoperable,” to add a note permitting a one-time increase in the Completion Time (CT) from 30 days to 45 days while specific compensatory measures are implemented to manage risk. The allowance for an extended completion time expires on October 10, 2021.

To complete its review of the inspection, the U.S. Nuclear Regulatory Commission (NRC) staff requests the below additional information.

On September 21, 2021, the NRC staff provided draft request for additional information (RAI) questions to SNC to make sure that the RAIs are understandable, the regulatory basis is clear, to ensure there is no proprietary information, and to determine if the information was previously docketed. The NRC staff is requesting that SNC would provide the RAI response commensurate with the emergency situation.

If you have any questions, you can contact me at 301-415-3100.

Sincerely,

John

REQUEST FOR ADDITIONAL INFORMATION (RAIs)

By letter dated September 20, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21264A003), the Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted a license amendment request (LAR) for the Edwin I. Hatch Nuclear Plant (Hatch), Unit 1. The proposed amendment would revise the Hatch, Unit 1 Technical Specification (TS) requirements of TS 3.7.2, “Plant Service Water (PSW) System and Ultimate Heat Sink (UHS).” Specifically, the proposed amendment would revise Limiting Condition for Operation (LCO) 3.7.2, Condition A, “One PSW pump inoperable,” to add a note permitting a one-time increase in the Completion Time (CT) from 30 days to 45 days while specific

compensatory measures are implemented to manage risk. The allowance for an extended completion time expires on October 10, 2021.

Regulatory Requirements

The regulation under Title 10 of the *Code of Federal Regulation*s (10 CFR), Part 50, Section 36(c)(2) requires that TSs contain LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met. Typically, the TSs require restoration of equipment in a timeframe commensurate with its safety significance, along with other engineering considerations. The regulation under 10 CFR 50.36(b) requires that TSs be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto.

In determining whether the proposed TS remedial actions should be granted, the U.S. Nuclear Regulatory Commission (NRC) staff applies the “reasonable assurance” standards of 10 CFR 50.40(a) and 50.57(a)(3). The regulation at 10 CFR 50.40(a) states that in determining whether to grant the licensing request, the Commission will be guided by, among other things, consideration about whether “the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in Part 20 of this chapter, and that the health and safety of the public will not be endangered.” Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” describes a risk-informed approach that includes deterministic considerations to support this reasonable assurance finding.

Probabilistic Risk Assessment (PRA)

PRA - RAI 1 – PRA Change Since 10 CFR 50.69 and NFPA 805 Reviews

The NRC has previously reviewed the Hatch internal events, internal flooding, internal fire, and seismic PRAs (IEPRA, IFPRA, FPRA, and SPRA, respectively) for determining their acceptability to support the Hatch National Fire Protection Association (NFPA) 805 program (ADAMS Accession Nos. ML18096A955 and ML19280C812), and the Hatch 10 CFR 50.69 License Amendment Request (ADAMS Accession Nos. ML18158A583 and ML19197A097). The NRC staff concluded that the information was acceptable for the application. In this emergency LAR, the licensee referred to the aforementioned LARs for discussion on PRA Technical Adequacy. In Regulatory Guide (RG) 1.200, Regulatory Position 4.2, licensees are expected to “address the need for the PRA model to represent the as-designed or as-built, as-operated plant.” Therefore, crediting previously reviewed analysis is appropriate as long as the previous technical conclusions reflect that the PRA model continues to reflect the as-built, as operated plant for the current amendment.

a. Describe any updates or potential upgrades made to the PRA models since the approval of the 10 CFR 50.69 and NFPA 805 programs.

b. If there were updates or potential upgrades made to the PRA, evaluate their impact to the current requested license amendment.

PRA - RAI 2 – Compensatory Measures License Condition

The RG 1.177, Tier 2 evaluation identifies which systems, structures, and components (SSCs), in combination with the component already out of service, could result in a risk significant configuration. The licensee presented a number of SSCs which have been identified as compensatory measures with increased importance during this outage. The SSCs that become more important are associated with the other plant service water (PSW) pumps, high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), the 1B Emergency Diesel Generator, and the Containment Hardened Vent.

If any of these SSCs become Inoperable during this additional 15-days to the CT, there is a potential that the configuration risk profile in the facility would exceed the acceptance criteria required for the requested outage period . Therefore,

a. Describe the licensee's plan to address any potential outages of the SSCs identified as a part of the Tier 2 evaluation.

b. Alternatively, propose a mechanism that avoids a risk significant configuration from an outage of the SSCs mentioned above.

c. In Attachment 4, section 2.1.3, Calculation Approach, the licensee described the addition of recovery rules. Describe which rules were applied and if the PSW pump 1C outage configuration has any impact to them.

PRA - RAI 3 – Facts and Observations (F&Os)

The licensee did not provide details on any open F&Os and disposition in the LAR. In Attachment 4, Section 1.5, the licensee stated that "all of the F&Os [for each hazard model] have been addressed." However, the licensee also stated there are "two open findings related to internal flooding documentation that do not impact the outcome of this assessment." RG 1.200, Regulatory Position 4.2 stated that NRC staff expects a licensee to discuss "the resolution of the peer review…findings and observations that are applicable to the parts of the PRA required for the application."

Provide details of any open F&Os and associated applicability to the results of this LAR.

PRA - RAI 4 – Common Cause Failure

In Attachment 4, Section 2 of the LAR, the licensee described its approach to adjusting for common cause failures. It provided a total random failure rate (Qt) for failure-to-start (FTS) and failure-to-run (FTR) of 1.79E-6/hour (hr) and 1.48E-3/hr, respectively. The NRC staff ran the Hatch Nuclear Plant SPAR model and produced incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP) values that were more conservative (higher) than that of the SNC model. Comparing the available hazards in SPAR and the SNC models revealed little differences between internal events but a large difference in the contribution of internal fires.

Recognizing the SNC Fire PRA model is best available information due to the NFPA 805 transition and related plant modifications being reflected in the licensee's model, the NRC staff requests that the licensee run the fire PRA model with the NRC's adjusted common-cause failures (CCF) for PSW Pump FTS and FTR using the following values:

Adjusted CCF FTS : 8.033 E-3
Adjusted CCF FTR : 2.077 E-3

Please provide the updated ICCDP and ICLERP estimates for all hazards (internal events, fire, internal flood, and seismic) using the OTMHM with the NRC's adjusted CCF FTS and CCF FTR probabilities.

Technical Specifications

Background

Hatch Technical Specification (TS) 1.3, "Completion Times" establishes the CT convention and provides guidance for its use. Hatch TS LCO 3.0.1 through 3.0.8 contain usage requirements for LCOs. Part of the NRC staff's review includes evaluation of the proposed TS change for conformance to the conventions and requirements contained in the existing TS to ensure a proposed change, once implemented, will continue to provide adequate assurance of public health and safety.

STSB - RAI 5

The current proposed text of the TS NOTE above the existing 30 day CT for Required Action A.1 states "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."

Please address the following aspects of the proposed NOTE:

5a) Given the current CT is 30 days and the plant remains in Condition A since entry in August and the current CT will expire September 25, please explain why 45 days was chosen for the NOTE instead of the alternative of stating the allowance in terms of number of days requested in excess of the current 30 day CT.

At the end of Section 2.5 on page E-8, compensatory measures are mentioned as they relate to the allowance: "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."

5b) Please provide a discussion of whether establishing the compensatory measures is a prerequisite to using the allowance before exceeding 30 days in the condition where the 1C pump is inoperable.

5c) Please provide a discussion of how operators would respond if any of the compensatory measures are found to be not implemented after commencing use of the allowance.

The current text of the NOTE states the allowance expires at 1620 EDT on October 10, 2021.

5d) Please provide a discussion of whether or not there is a need for text explicitly stating the allowance would no longer apply after restoration of the 1C pump.

STSB - RAI 6

Section 4.1 of the request, on page E-11 of the application states: “The proposed amendment does not alter the remedial actions or shutdown requirements required by 10 CFR 50.36(c)(2)(i).” The NRC staff evaluates acceptability of remedial actions based on the actions required as well as time allowed to complete the actions. In this case, the staff believes both the action and time component of remedial actions would be altered.

Please provide a discussion explaining how the allowance increases time allowed in MODE 1with an inoperable 1C PSW pump contingent on certain compensatory measures being in place.

Plant Systems

SCPB – RAI 7

The second key principle of RG 1.174 relates to evaluation of defense-in-depth, which includes consideration of the potential for common cause failures. Section 2.1, “Emergency Circumstances,” of the Enclosure to the LAR described that operators found the 1C PSW pump to have excessive vibration on August 26, 2021, and subsequently shut down the pump and declared it inoperable. Troubleshooting identified the 1C PSW pump had the following conditions:

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

Please provide an assessment of potential causes of these conditions, including common maintenance practices (e.g., fastener torque procedure and practices, shaft alignment procedures and practices, and adequacy of post-maintenance testing applicable to the PSW pumps), condition monitoring practices (e.g., type, frequency and acceptance criteria for in-service tests), material or component degradation, and design defects. Also, please assess the applicability of these factors in presenting a challenge to the continued operability of the other Hatch Unit 1 PSW pumps currently considered operable. Describe any Southern Nuclear Operating Company operating experience indicating a similar failure mode on deep-draft water pumps involving loosened fasteners or separated components and the identified cause.

SCPB – RAI 8

The second key principle of RG 1.174 relates to evaluation of defense-in-depth, which includes avoidance of over-reliance on compensatory measures. Section 3.3, “Compensatory Measures,” of the Enclosure to the LAR described several actions involving classification of components as “protected” and deferring preventive maintenance on FLEX pumps. Please explain (1) the meaning of “protected” as it relates to component maintenance, (2) the risk-informed basis for designating only the 1A PSW pump rather than all operable PSW pumps as “protected,” and (3) the TS required surveillances expected to be performed on the 1B diesel generator and the standby service water pump during the proposed extended completion time and their effect on availability. Also, please describe (1) the modeling of the FLEX equipment in the risk assessment, (2) the current operational status and reliability experience with the FLEX equipment modeled to compensate for PSW system failures (e.g, portable generators and cooling water pumps), and (3) the expected effect of deferred maintenance on the reliability of this FLEX equipment.

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