Southern Nuclear

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

Edwin I. Hatch Nuclear Plant – Units 1 and 2
Application to Revise Technical Specifications
Surveillance Requirements to Increase Safety/Relief Valves Setpoint
Response to Request for Additional Information

Ladies and Gentlemen:

On April 19, 2024, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) for Edwin I. Hatch Nuclear Plant (Hatch), Units 1 and 2. The proposed LAR revises Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to increase the nominal mechanical relief setpoints for all safety/relief valves (S/RVs) of the reactor coolant system (RCS) nuclear pressure relief system. The proposed changes will reduce the potential for S/RV pilot leakage. As a result of the increased S/RV setpoints, the LAR also proposes to change SR 3.1.7.7 to increase the minimum Standby Liquid Control pump discharge pressure accordingly.

On July 26, 2024, the U.S. Nuclear Regulatory Commission (NRC) staff determined that additional information was needed to complete its review of the LAR. The SNC response to the staff request for additional information (RAI) is enclosed.

This letter contains no NRC commitments.

The conclusions of the No Significant Hazards Consideration Determination Analysis and Environmental Consideration contained in the original LAR have been reviewed and are unaffected by this RAI response.

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 to the designated state official.

If you should have any questions regarding this submittal, please contact Ryan Joyce at 205.992.6468.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 23rd day of August 2024.

Jamie M. Coleman
Regulatory Affairs Director
Southern Nuclear Operating Company

rmj/efb/cbg

Enclosure: Response to Request for Additional Information

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

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Enclosure to NL-24-0313

Response to Request for Additional Information

NRC Request For Additional Information

NRC Question 1

10 CFR 50.36(c)(3), "Surveillance requirements," states that surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Section 3.0 of Enclosure 1 to the LAR discusses GE Nuclear Energy technical report NEDC-32041P, "Safety Review for Edwin I. Hatch Nuclear Power Plant, Units 1 and 2, Updated Safety/Relief Valve Performance Requirements," Rev. 2, April 1996, which provided a detailed justification for an upper value mechanical S/RV relief setpoint as high as 1,195 psig, with one S/RV inoperable and at least 50 psi margin to the ASME BPV Code upset limit (1,375 psig). Section 2.2 of NEDC-32041P states that this margin allows for variations in the peak vessel pressure which were calculated for past cycles and may be predicted for future fuel cycles. This implies that some of the 50 psi margin may be used, and therefore, peak pressure could potentially be larger than 1,325 psig. However, the Hatch TS Safety Limit in TS 2.1.2 states that reactor steam dome pressure shall be ≤ 1,325 psig.

Please confirm that the steam dome pressure safety limit is not exceeded in a) the analysis performed for the subject LAR and b) as part of the cycle-specific reload licensing analyses for future cycles.

SNC Response to Question 1

The peak vessel pressure values reported in NEDC-32041P using the ODYN code are the maximum pressure anywhere in (i.e., at the bottom of) the reactor pressure vessel. The upper limit S/RV setpoint of 1,195 psig was selected analytically in NEDC-32041P to maintain, at that time, approximately 50 psi margin between the calculated peak vessel (bottom) pressure and the ASME limit of 1,375 psig. The Hatch upper limit S/RV setpoint of 1,195 psig is not being changed as a part of this LAR.

The Hatch Technical Specification Safety Limit (SL) 2.1.2 is that the *steam dome* pressure shall be ≤ 1,325 psig. The Technical Specification Bases for SL 2.1.2 states that 1,325 psig in the reactor steam dome is equivalent to 1,375 psig at the lowest elevation of the RCS.

The highest peak vessel pressure result in NEDC-32041P (Table 3-2) is 1,321 psig with one SRV out-of-service (i.e., 54 psi of margin to the ASME limit). Given that this is the peak vessel pressure at the bottom of the vessel, the corresponding dome pressure would be lower than 1,321 psig and necessarily lower than the TS SL dome pressure of 1,325 psig.

Each cycle for Hatch-1 and Hatch-2, as part of the reload licensing analysis process, the overpressurization analysis is performed using the approved methodology with an S/RV opening setpoint of 1,195 psig and one SRV out-of-service. The analysis confirms that both the upper-95/95 peak pressure in the steam dome will be ≤ 1,325 psig (TS SL) and the overall peak pressure (i.e., at the bottom of the vessel) will be ≤ 1,375 psig (TS Bases and ASME limit).

NRC Question 2

The NRC issued Generic Letter (GL) 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” (June 28, 1989) to extend the scope of the motor-operated valve (MOV) testing program in Bulletin 85-03 and its Supplement 1 to all safety-related MOVs in nuclear power plants. The NRC issued seven supplements to GL 89-10 as MOV issues were identified during its implementation by nuclear power plant licensees. As a follow-up to the GL 89-10 programs, the NRC issued GL 96-05, “Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,” (September 18, 1996) to request that licensees verify on a periodic basis that safety-related MOVs continue to be capable of performing their safety functions within the current licensing bases of the facility.

Historically, MOV inservice testing was performed under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, until the preparation of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), usually without fluid pressure or flow in the lines (referred to as static testing). Based on operating and research experience, and the results of the GL 89-10 and 96-05 programs, the NRC conducted rulemaking to require in 10 CFR 50.55a(b)(3)(ii) as a compliance backfit that licensees must establish a program to ensure that MOVs continue to be capable of performing their design basis safety functions in addition to the ASME OM Code testing requirements. The purpose of the rulemaking is to periodically confirm that MOVs will perform their safety functions under design-basis conditions (i.e., under pressure and flow conditions) and, if possible, to identify the causes of MOV failures.

Section 3.0 of Enclosure 1 to the LAR states “The impacts on MOVs due to the potential for increased reactor vessel and system pressure as a result of the increase in the S/RV nominal opening setpoint are evaluated in accordance with the Generic Letter 89-10 requirements as part of the SNC design process.” However, there are no results provided of any evaluation.

Please provide a summary of SNC's evaluation of MOVs in the high-pressure systems (HPCI and RCIC) to confirm they will remain operable given the increased S/RV setpoint pressure.

SNC Response to Question 2

Design Change Package (DCP) SNC1523201 is an in-progress design change which would be responsible for both implementing the S/RV setpoint increase requested by the subject LAR and evaluating all ancillary systems and components affected by the setpoint increase, including all affected MOVs within the systems.

SNC follows the industry standard design process (IP-ENG-001) which requires a Design Attribute Review (DAR) be completed for all modifications. The DAR is “a review performed during development of an Engineering Change to determine applicable or impacted engineering disciplines, engineering programs and stakeholders from other departments, areas or programs.” The scoping questions asked by the DAR ensure MOVs are identified as affected. IP-ENG-001 along with site specific procedures have requirements that ensure all necessary changes to the affected systems and components are implemented prior to or concurrent with the overall change being made, which in this case, is the S/RV setpoint increase. All physical modifications required by the design must be implemented before the S/RVs can be returned to service at the increased mechanical setpoint.

The on-going design change will identify the necessary calculations, documentation, and/or valve setup changes necessary to implement the new S/RV setpoint while also ensuring the impacted MOVs will maintain sufficient margin to meet GL 96-05 requirements as well as conforming to testing requirements as specified in the generic letter. This will also include making any necessary physical modifications to the valves prior to changing the S/RV setpoints.

Currently, the design change has not progressed to a point where an in-depth evaluation of affected MOVs has been performed. Although formal evaluations have not yet been performed, the design change process will ensure the continued operability of all affected systems and components, including impacted MOVs, at the increased S/RV setpoints.