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PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

April 19, 2024

Docket Nos.: 50-321
50-366

NL-24-0026
10 CFR 50.90

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

Ladies and Gentlemen:

Pursuant to the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations*, Southern Nuclear Operating Company (SNC) hereby requests amendments to renewed facility operating licenses DPR-57 and NPF-5 to revise the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2, respectively. The proposed changes would revise 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 (NPRS). The proposed changes will reduce the potential for S/RV pilot leakage. As a result of the increased S/RV setpoints, a change is proposed to SR 3.1.7.7 to increase the minimum Standby Liquid Control pump discharge pressure accordingly.

Enclosure 1 to this letter provides a description and assessment of the proposed changes. Attachment 1 provides the existing TS pages marked to show the proposed changes. Attachment 2 provides revised (clean) TS pages. Attachment 3 provides existing TS Bases pages marked to show the proposed changes for information only. Attachment 4 contains a GE Hitachi Nuclear Energy (GEH) proprietary report which details safety analyses performed in support of the proposed change. Pursuant to 10 CFR 2.390(a)(4), SNC requests that the proprietary information be withheld from public disclosure. In accordance with 10 CFR 2.390(b)(1), an affidavit attesting to the proprietary nature of the enclosed information and requesting withholding from public disclosure is included with Attachment 4. Attachment 5 provides the same GEH information with the proprietary portions removed and is provided for public disclosure.

These changes would be implemented during a scheduled refueling outage on each unit. The next Unit 2 refueling outage is scheduled for February, 2025, and the next Unit 1 refueling outage is scheduled for February, 2026. Therefore, to support the upcoming refueling outages and to provide adequate time for outage preparation, SNC requests that the NRC review and approve the amendments no later than December 13, 2024, with implementation prior to startup from the respective refueling outages.

This letter contains no NRC commitments.

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 19th day of April 2024.

Jamie Coleman

Jamie M. Coleman
Regulatory Affairs Director
Southern Nuclear Operating Company

rmj/efb/cbg

Enclosure: 1) Description and Assessment of the Proposed Changes

Attachments:
1) Proposed Technical Specification Changes (Mark-up)
2) Revised Technical Specification Pages
3) Proposed Technical Specifications Bases Changes (Mark-up) – For Information Only
4) GEH Affidavit and Proprietary GEH Report NEDC-34126P, Revision 0
5) Non-Proprietary GEH Report NEDO-34126, Revision 0

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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NL-24-0026

Enclosure 1

Description and Assessment of the Proposed Changes

1.0 SUMMARY DESCRIPTION

Southern Nuclear Operating Company (SNC) proposes to revise Edwin I. Hatch Nuclear Plant (HNP) Unit 1 and Unit 2 Technical Specifications (TS) to increase the nominal mechanical relief setpoints for each unit's 11 safety/relief valves (S/RVs) of the reactor coolant system (RCS) nuclear pressure relief system (NPRS) from 1150 psig to 1160 psig. Changes are proposed to Surveillance Requirement (SR) 3.4.3.1 to increase these setpoints. These changes do not alter the minimum number of S/RVs required to be operable, nor do they alter the allowable as-found or as-left tolerances as a percentage of the nominal setpoint. As a result of the increased S/RV setpoints, a change is proposed to SR 3.1.7.7 to increase the minimum Standby Liquid Control pump discharge pressure accordingly.

In support of the proposed changes, GE Hitachi Nuclear Energy (GEH) prepared and issued GEH report NEDC-34126P, "Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Safety/Relief Valve Setpoint Increase," Revision 0, dated March 2024. A proprietary copy of this report is provided in Attachment 4 and a non-proprietary version is provided in Attachment 5. The results of the evaluations in the GEH report determined that the impacts of the setpoint changes are acceptable.

Unless noted otherwise, the information provided throughout this License Amendment Request (LAR) is applicable to both Unit 1 and Unit 2. Additionally, "setpoint" throughout this LAR refers to the SR 3.4.3.1 mechanical relief setpoint of the NPRS S/RVs.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the NPRS, the size and number of S/RVs are selected such that peak pressure in the nuclear steam system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The NPRS for each unit includes 11 S/RVs, all of which are located on the main steam lines within the drywell between the reactor vessel and the first isolation valve. In the safety mode of the S/RVs, the spring-loaded pilot valve opens when steam pressure at the valve inlet expands the bellows to the point that the bellows force overcomes the force holding the pilot valve closed. Opening the pilot valve allows steam to pass to the second stage operating piston which causes the second stage disc to open. This vents the chamber over the main valve disc to the downstream side of the valve, which causes a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the ASME Boiler and Pressure Vessel Code requirement. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The Standby Liquid Control (SLC) System provides the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. Additionally, the SLC system provides sufficient buffering agent to maintain the suppression pool pH at or above 7.0 following a Design Basis Loss of Coolant Accident involving fuel damage.

The SLC System consists of a storage tank, two positive displacement pumps, two relief valves (one on discharge of each pump), two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor. The SLC System is manually initiated from the Control Room as directed by the emergency operating procedures and provides an independent, redundant reactivity control system to shut down the reactor in the unlikely event that the Control Rod Drive System fails to insert control rods during scram conditions. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for the various reactivity effects that could occur during plant operations.

2.2 Current Technical Specifications Requirements

Limiting Condition for Operation (LCO) 3.4.3 for both units requires the safety function of 10 of 11 S/RVs to be Operable. The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

SR 3.4.3.1 requires verification that the safety function lift setpoints of the S/RVs are 1150 ± 34.5 psig. The safety function of the S/RV lift settings is demonstrated by bench testing performed on S/RV pilot valves that are removed during shutdown in accordance with the Inservice Testing Program. The lift setting pressure must correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint tolerance is ± 3% (34.5 psig) for operability; however, the valves are reset to a ± 1% tolerance during the Surveillance to allow for drift.

LCO 3.1.7 for both units requires two SLC subsystems to be operable. The operability of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the operability of the pumps and valves.

SR 3.1.7.7 requires verification that each SLC pump develops a flow rate greater than or equal to 41.2 gpm at a discharge pressure greater than or equal to 1232 psig.

2.3 Reason for Proposed Change

The NPRS is robust, but S/RV leakage has occurred during plant operation. Increasing the nominal mechanical relief setpoints will increase the simmer margin (i.e., the difference between the S/RV setpoints and the vessel steam dome pressure), thereby potentially reducing S/RV pilot leakage which may occur during a typical operating cycle.

2.4 Description of Proposed Change

The proposed change revises SR 3.4.3.1 for both units to change the 1150 psig setpoint to 1160 psig. The setpoint tolerance (± 3% of the setpoint value), currently 34.5 psig, is revised to 34.8 psig.

Additionally, SR 3.1.7.7 for both units is proposed to be revised to change the minimum SLC pump discharge pressure from 1232 psig to 1251 psig.

Associated changes are proposed to the Unit 1 and Unit 2 TS Bases. The GEH report is added as a reference in the TS Bases for justification of the S/RV safety lift settings.

3.0 TECHNICAL EVALUATION

On October 7, 1996, a LAR for HNP Units 1 and 2 was submitted to the Nuclear Regulatory Commission (NRC) to increase the nominal mechanical relief setpoints for all NPRS S/RVs to their current nominal value of 1150 psig (Reference 1). This LAR was subsequently approved by the NRC on March 21, 1997 (Reference 2). The NRC safety evaluation was based on the evaluations documented in technical report NEDC-32041P, Revision 2, as provided in the 1996 LAR. This technical report provided a detailed justification for an upper value mechanical S/RV relief setpoint as high as 1195 psig, with one S/RV inoperable and at least 50 psi margin to the ASME code upset limit (1375 psig). The 1195 psig upper limit (UL) established by NEDC-32041P bounds the current nominal setpoint including a ±3% drift tolerance.

Hatch currently performs cycle-specific analyses that confirm vessel overpressure margin is maintained assuming S/RV opening at the UL of 1195 psig. The UL value of 1195 psig continues to bound the proposed nominal setpoint plus maximum allowable drift tolerance (1160 + 34.8 psig) such that the cycle-specific reload licensing analyses demonstrating overpressure protection are unaffected by this change.

GEH report NEDC-34126P provides additional evaluations of the following non-cycle-specific areas potentially affected by the proposed change:

- High Pressure System Performance (High Pressure Coolant Injection (HPCI)/ Reactor Core Isolation Cooling (RCIC) operation)
- Emergency Core Cooling System (ECCS)/Loss of Coolant Accident (LOCA) performance
- Containment Evaluation (Anticipated Transients Without Scram (ATWS), Design Basis Accident (DBA) LOCA, Small Steam Line Break (SSLB) for Equipment Qualification (EQ), Appendix R, and Station Blackout (SBO))
- ATWS Mitigation

S/RV discharge piping loads and Standby Liquid Control (SLC) System performance were also reassessed for the effects of increasing the nominal S/RV setpoint.

The following is a brief description of the evaluations discussed in Attachment 4, along with the assessments of S/RV discharge piping loads and SLC System performance:

<u>ECCS/LOCA Evaluation</u>
Section 3.0 of GEH NEDC-34126P discusses the effect of the S/RV setpoint change on the peak cladding temperatures for the HNP ECCS LOCA. Hatch Units 1 and 2 are licensed to the TRACG-LOCA best estimate plus uncertainty ECCS/LOCA evaluation methodology. Using the same approved TRACG-LOCA methodology, an analysis was performed using representative limiting break locations to determine the effect of increasing the S/RV opening setpoint by running the break spectra for those break locations. This analysis determined that the licensing basis ECCS/LOCA results are not affected by increasing the S/RV opening setpoint nominal value from 1150 psig to 1160 psig.

<u>High Pressure System Performance</u>
Section 4.0 of GEH NEDC-34126P discusses the performance of the HPCI and RCIC Systems with the increase in the S/RV setpoints. Operation at reactor pressures up to the UL is within the design limits for system piping, pumps, and turbines for the HPCI and RCIC systems. 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. The HPCI and RCIC pumps are capable of delivering rated system flow with vessel pressures at the UL value of 1195 psig.

Containment Evaluation
Section 5.0 of GEH NEDC-34126P discusses effects of the proposed increase in S/RV setpoints on containment-related evaluations, which include ATWS, DBA LOCA, SSLB for EQ, Appendix R, and SBO. The evaluations were performed with the same methodologies as the current bases for these events.

- ATWS – The evaluation performed for ATWS demonstrated that the peak wetwell pressure and temperature with the proposed S/RV setpoint change were equal to or bounded by the current analysis of record.
- DBA LOCA – The evaluation determined that both long-term and short-term DBA LOCA analyses are unaffected by the proposed S/RV setpoint increase from 1150 to 1160 psig.
- SSLB for EQ – The SSLB containment analysis demonstrated that the S/RV setpoint increase results in negligible changes in the drywell temperature curves for the various break sizes. As such, there is negligible effect on HNP Units 1 and 2 EQ profile.
- Appendix R – Hatch Units 1 and 2 are now licensed to NFPA 805 for fire protection. However, the deterministic Appendix R containment response evaluation was conservatively assessed for impact. The effect on the suppression pool temperature response due to S/RV setpoint increase was determined to be negligible and, in turn, the effect on containment temperature and pressure are negligible. It was concluded that there is negligible effect on the Appendix R containment response from increasing the nominal S/RV setpoint.
- SBO – The station blackout event is also an RPV isolation and non-break event similar to Appendix R. The applicable discussion and conclusion for Appendix R is also applicable to SBO. Thus, there is negligible effect on the SBO response from increasing the S/RV setpoint.

ATWS Mitigation Capability
Section 6.0 of GEH NEDC-34126P discusses the S/RV setpoint increase impacts on ATWS acceptance criteria compliance for limiting ATWS events. The limiting ATWS events of Main Steam Isolation Valve Closure (MSIVC) and Pressure Regulator Failure Open (PRFO) were analyzed to demonstrate compliance with the following:

- ASME Service Level C Pressure Limit (1500 psig)
- Containment Pressure Design Limit (plant-specific, see Attachment 4)
- Suppression Pool Temperature Design Limit (plant-specific, see Attachment 4)
- 10 CFR 50.46 PCT Limit (<2200F)
- 10 CFR 50.46 Local Cladding Oxidation Thickness Limit (<17%)

Based on the analysis results, all ATWS acceptance criteria are met for the S/RV setpoint increase from 1150 psig to 1160 psig.

SLC System Performance

The SLC system required pump discharge pressure is based on the limiting peak pressure at the SLC injection location (lower plenum injection) after SLC System initiation from a MSIV closure event at the beginning of an operating cycle. With an increase in S/RV setpoints to 1160 psig and a SLC System initiation time of 130.6 seconds, the resulting pressure at the SLC System injection location is 1218 psia (1203.3 psig). SLC System losses were determined to be approximately 47 psi. Using the lower plenum pressure, the required SLC pump discharge pressure will become 1251 psig (1203.3 + 47). This value of 1251 psig is the proposed SR 3.1.7.7 minimum SLC pump discharge pressure.

The SLC pumps are positive displacement pumps, which deliver a constant flow rate regardless of discharge pressure. The pump motors are 40 hp, which are adequate for the pressure increase. The system design pressure is adequate for the increase in operating pressure.

The SLC System pump discharge relief valve setpoint margin is based on the discharge pressure during an ATWS. NRC Information Notice (IN) 2001-13 identifies the need to include a margin of 75 psi to prevent inadvertent actuation of the SLC System relief valves. This margin accounts for pressure pulsations from the positive displacement pumps and tolerance for the SLC System discharge relief valves. The maximum RPV lower plenum pressure without SLC System relief valves lifting is the SLC System relief valve setpoint (1400 psig) minus the 100 psi margin to prevent inadvertent actuation, minus the SLC System piping losses (47 psi). This results in a maximum RPV lower plenum pressure without the SLC System relief valve lifting of 1253 psig (1400 – 100 - 47).

As a result of the proposed S/RV setpoint increase, the updated peak pressure at the SLC injection location (lower plenum) after SLC injection is 1203.3 psig (1218 psia). Therefore, the additional pressure margin to relief valve lift is 49.7 psi (1253 psig – 1203.3 psig). This represents an additional 74.7 psi (49.7 + 25) margin above the 75 psi margin in NRC IN 2001-13. Based on this review, the current SLC System relief valves and their associated setpoints are acceptable for the proposed increase in S/RV setpoints.

S/RV Discharge Line Loads

SNC performed an assessment of the impact of increasing the S/RV nominal setpoint to 1160 psig on the S/RV discharge line loads for HNP Units 1 and 2. The updated analyses for both units demonstrated that the current configuration of all 11 S/RV discharge line piping portions located within the vent pipes and torus meet ASME Code requirements for all load combinations.

Conclusion

Evaluations have been performed which consider the consequences of the various transients and accidents with the increased setpoints. The evaluations also analyze the impact on SLC and ECCS performance, including HPCI and RCIC. The conclusions of these evaluations have shown no significant increase in consequences of an accident with the increased S/RV setpoints.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, "Technical specifications"
Regulation 10 CFR 50.36, "Technical specifications," provides the requirements for the content required in the TS. As stated in 10 CFR 50.36, the TSs include, among other things, LCOs and SRs 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. As described above, the SRs are proposed to be updated to assure that the facility operation is within safety limits.

ASME Boiler and Pressure Vessel Code
The ASME Boiler and Pressure Vessel Code requires that each vessel designed to meet Section III be protected from overpressure. The code allows a peak allowable pressure of 110% of vessel design pressure. The NPRS SR/Vs are designed and manufactured in accordance with ASME Boiler and Pressure Vessel Code Section III, 1968 Edition with Addenda through 1970. The evaluations described in Section 3.0 above conclude that the proposed TS changes will continue to assure that the design requirements associated with the S/RVs and their associated functions are met.

4.2 Precedent

Reference 1 provides a previous example of a similar license amendment approved by the NRC for HNP to increase the nominal mechanical relief setpoints for all NPRS S/RVs to their current nominal value of 1150 psig. References 3 and 4 provide examples of other industry license amendments involving S/RV setpoint and setpoint tolerance changes which involve similar technical analyses to those used for the proposed HNP changes.

4.3 No Significant Hazards Consideration Determination Analysis

Southern Nuclear Operating Company (SNC) proposes to revise Edwin I. Hatch Nuclear Plant (HNP) Unit 1 and Unit 2 Technical Specifications (TS) to increase the nominal mechanical relief setpoints for each unit's 11 safety/relief valves (S/RVs) of the reactor coolant system (RCS) nuclear pressure relief system (NPRS) from 1150 psig to 1160 psig. Changes are proposed to Surveillance Requirement (SR) 3.4.3.1 to increase these mechanical relief setpoints. As a result of the increased S/RV setpoints, a change is proposed to SR 3.1.7.7 to increase the minimum Standby Liquid Control pump discharge pressure accordingly.

SNC has evaluated if a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The S/RVs serve to mitigate postulated transients and accidents; the proposed changes do not alter the function or mode of operation of the S/RVs. The probability of an operable or an inoperable S/RV inadvertently opening or failing to open or close is not affected by these changes. The proposed change does not alter the safety function of the valves. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions and no changes to existing structures, systems, or components. Therefore, the probability of an accident is not increased. Evaluations have been performed which consider the consequences of the various transients and accidents with the increased setpoints. The evaluations also analyze the impact on ECCS performance, including HPCI and RCIC. The conclusions of these evaluations have shown no significant increase in consequences of an accident with the increased S/RV setpoints.

Therefore, the proposed change 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 kind of accident from any accident previously evaluated?

   Response: No

   Revising the nominal S/RV setpoint only changes when the S/RV opens in its safety mode; the operation of the S/RV and any other existing equipment is not altered. The impact on the operation and design of other systems and components has been evaluated, including ECCS and SLC. The proposed change does not affect the manner in which the NPRS is operated; therefore, there are no new failure mechanisms for the NPRS. The proposed change does not change the safety function of the valves. There is no alteration to the parameters within which the plant is normally operated. As a result, no new operating or failure modes are being introduced. Thus, these changes do not contribute to a new or different type of accident.

   Therefore, the proposed change does not create the possibility of a new or different kind of 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 established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not modify the safety limits or setpoints at which protective actions are initiated and does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. The change in S/RV mechanical lift setpoint was evaluated relative to the applicable safety system settings and found to remain acceptable. The proposed changes were evaluated against peak clad temperature limits, ECCS operation, ASME Code overpressurization limits, and containment design limits. No significant reduction in the margin of safety was identified in the evaluations performed.

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

Based on the above, SNC concludes that the proposed change presents no 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 above, (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.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion 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.

6.0 REFERENCES

1. Letter from Georgia Power Company to NRC, "Edwin I. Hatch Nuclear Plant Request to Revise Technical Specifications: Safety/Relief Valve Setpoint Change," dated October 7, 1996 (ADAMS Accession No. ML20128M857)
2. Letter from NRC to Georgia Power Company, "Issuance of Amendments – Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TAC Nos. M96752 and M96753)," dated March 21, 1997 (ADAMS Accession No. ML013030262)
3. Letter from Entergy Nuclear Operations, Inc. to NRC, "Proposed License Amendment to Technical Specifications: Revised Technical Specification for Setpoint and Setpoint Tolerance Increases for Safety Relief Valves (SRV) and Spring Safety Valves (SSV), and Related Changes," dated March 15, 2010 (ADAMS Accession ML100770450)
4. Letter from Exelon Generation Company, LLC to NRC, "License Amendment Request to Revise the Technical Specification (TS) Surveillance Requirement (SR) 3.4.4.1 to Revise the Lower Setpoint Tolerances for Safety/Relief Valves (S/RVs)," dated February 27, 2018 (ADAMS Accession ML18058A257)

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Attachment 1

Proposed Technical Specification Changes (Mark-up)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ ~~1232~~ 1251 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	In accordance with the Surveillance Frequency Control Program AND Once within 24 hours after pump suction piping temperature is restored within the Region A limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 60.0 atom percent B-10.	Prior to addition to SLC tank

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the S/RVs are as follows: Number of S/RVs — Setpoint (psig) 11 — ~~1150~~ 1160 ± ~~34.5~~34.8 Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual and power operated 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
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ ~~1232~~ 1251 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	In accordance with the Surveillance Frequency Control Program AND Once within 24 hours after pump suction piping temperature is restored within the Region A limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 60.0 atom percent B-10.	Prior to addition to SLC tank

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the S/RVs are as follows: Number of S/RVs — Setpoint (psig) 11 — ~~1150~~ 1160 ± ~~34.5~~34.8 Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

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Attachment 2

Revised Technical Specification Pages

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1251 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 24 hours after pump suction piping temperature is restored within the Region A limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 60.0 atom percent B-10.	Prior to addition to SLC tank

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the S/RVs are as follows: Number of S/RVs — Setpoint (psig) 11 — 1160 ± 34.8 Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual and power operated 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
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1251 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	In accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 24 hours after pump suction piping temperature is restored within the Region A limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 60.0 atom percent B-10.	Prior to addition to SLC tank

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the S/RVs are as follows: Number of S/RVs — Setpoint (psig) 11 — 1160 $\pm$ 34.8 Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

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Attachment 3

Proposed Technical Specifications Bases Changes (Mark-up) – For Information Only

BASES (continued)

ACTIONS

A.1 and A.2

With 1 SR/V inoperable, no action is required, because an analysis demonstrated that the remaining 10 SR/Vs are capable of providing the necessary overpressure protection. (See Ref. 5.)

With two or more S/RVs inoperable, a transient may result in the violation of the ASME Code limit on reactor pressure. The plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the S/RVs will open at the pressures assumed in the safety analysis of Reference ~~1~~5. The demonstration of the S/RV safety lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the INSERVICE TESTING PROGRAM. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

REFERENCES

1. FSAR, Appendix M.

2. Unit 2 FSAR, Chapter 15.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

4. NEDC-32041P, "Safety Review for Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Updated Safety/Relief Valve Performance Requirements," April 1996.

(continued)

BASES (continued)

REFERENCES (continued)

5. GEH Report NEDC-34126P, Rev. 0, "Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Safety/Relief Valve Setpoint Increase," March 2024.

BASES

BACKGROUND (continued)

is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to ~~1185~~ 1195 psig). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using a "keep fill" system (jockey pump system). The HPCI System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water. Therefore, HPCI does not require a "keep fill" system.

The ADS (Ref. 4) consists of 7 of the 11 S/RVs. It is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (CS and LPCI), so that these subsystems can provide coolant inventory makeup. Each of the S/RVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures (150 psig to ~~1185~~ 1195 psig). Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating

(continued)

BASES

APPLICABILITY (continued)

from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With 1 S/RV inoperable, no action is required, because an analysis demonstrated that the remaining 10 SR/Vs are capable of providing the necessary overpressure protection. (See Reference 4.)

With two or more S/RVs inoperable, a transient may result in the violation of the ASME Code limit on reactor pressure. The plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the S/RVs will open at the pressures assumed in the safety analysis of Reference ~~1~~5. The demonstration of the S/RV safety lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the INSERVICE TESTING PROGRAM. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

The Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Supplement 5A.

2. FSAR, Section 15.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

4. NEDC-32041P, "Safety Review for Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Updated Safety/Relief Valve Performance Requirements," April 1996.

5. GEH Report NEDC-34126P, Rev. 0, “Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Safety/Relief Valve Setpoint Increase,” March 2024.

BASES

BACKGROUND (continued)

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

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (162 psid to ~~1200~~ 1210 psid, vessel to pump suction). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using a "keep fill" system (jockey pump system). The HPCI System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water. Therefore, HPCI does not require a "keep fill" system.

The ADS (Ref. 4) consists of 7 of the 11 S/RVs. It is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (CS and LPCI), so that these subsystems can provide coolant inventory makeup. Each of the S/RVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves.

(continued)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS), RPV WATER INVENTORY CONTROL, AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures (150 psig to ~~1154~~ 1195 psig). Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating

(continued)

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

NL-24-0026

Attachment 5

Non-Proprietary GEH Report NEDO-34126, Revision 0

GE Hitachi Nuclear Energy

NEDO-34126
Revision 0
March 2024

Non-Proprietary Information

Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Safety/Relief Valve Setpoint Increase

INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-34126P, Revision 0, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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Revision Summary

Revision	Required Changes to Achieve Revision
0	Initial release.

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Acronyms and Abbreviations

Short Form	Description
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BOC	Beginning of Cycle
CST	Condensate Storage Tank
DBA	Design Basis Accident
DIR	Design Input Request
DW	Dry Well
ECCS	Emergency Core Cooling System
EHC	Electro-Hydraulic Control
EOC	End of Cycle
EQ	Equipment Qualification
GEH	GE-Hitachi Nuclear Energy Americas LLC
HCTL	Heat Capacity Temperature Limits
HNP	Hatch Nuclear Plant
HPCI	High Pressure Coolant Injection
LHGR	Linear Heat Generation Rate
LLS	Low-Low-Set
LOCA	Loss-of-Coolant Accident
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
NFI	New Fuel Introduction
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature

Short Form	Description
PRFO	Pressure Regulator Failure Open
psig	Pounds per square inch gauge
PUSAR	Power Uprate Safety Analysis Report
RCIC	Reactor Core Isolation Cooling
rpm	Revolutions per Minute
RPV	Reactor Pressure Vessel
SBO	Station Black Out
SLCS	Standby Liquid Control System
SRV	Safety/Relief Valve
SSLB	Small Steam Line Break
U1	Unit 1
U2	Unit 2
WW	Wet Well

1.0 Introduction

1.1 Purpose

The purpose of this evaluation is to address an increase of the safety relief valve (SRV) opening setpoint nominal value from 1150 psig to 1160 psig to provide operational margin and reduce potential leakage. Upon successful evaluation, the Hatch Nuclear Plant (HNP) can use this document to support a license amendment request for increasing the nominal setpoint.

2.0 Analysis Approach

2.1 Discussion of Analyses

In order to address the increase of the SRV opening setpoints, the following evaluations are required.

- Loss of Coolant Accident (LOCA)
- High Pressure System Performance
- Containment Performance
- Anticipated Transients Without Scram (ATWS)

Upon successfully addressing these areas for the new setpoint, HNP can use this document to support a license amendment request for increasing the nominal setpoint.

3.0 TRACG LOCA Evaluation

Increasing the SRV opening setpoint nominal value from 1150 psig to 1160 psig was analyzed for the HNP Emergency Core Cooling System (ECCS) Loss of Coolant Accident (LOCA) to determine its effect on the analysis of record in Reference 1. The same method as listed in Reference 1 was followed for this analysis with the input change documented in Reference 2. The analysis was done by selecting representative limiting break locations in Reference 1 to determine the effect of increasing the SRV opening setpoint by running the break spectra for those break locations. [[

]] The licensing basis results reported in Reference 1 are not affected by increasing the SRV opening setpoint nominal value from 1150 psig to 1160 psig and, therefore, remain valid for HNP.

4.0 High Pressure System Performance

High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) performance were evaluated for SRV setpoint drift to the upper limit value of 1195 psig. Operation at the upper limit provides a greater challenge to the HPCI and RCIC piping, pumps, and turbines than SRVs at the new nominal setpoints. These evaluations assure satisfaction of performance requirements for operation at both the upper limit and at the new nominal setpoints because operation at the upper limit bounds operation at the proposed nominal setpoints.

Both HPCI and RCIC systems are important in mitigating actual reactor vessel isolation and loss of feedwater events, even though HPCI and RCIC systems may not be modelled explicitly in design-basis LOCA analyses.

4.1 Effect of Higher SRV Setpoints on HPCI and RCIC Performance

Analyses indicate that operation at reactor pressures up to the upper limit is within design limits for system piping, pumps, and turbines for the HPCI and RCIC systems. Southern Nuclear Operating Company should verify compliance with Nuclear Regulatory Commission (NRC) Generic Letter 89-10 (Reference 3) requirements for valves in each of these systems. The HPCI and RCIC pumps are capable of delivering rated system flow with vessel pressures at the upper limit value of 1195 psig. Based on the HPCI and RCIC pump performance curves, the turbine speed required to deliver rated flow for each of these systems with reactor pressure at the upper limit are [[ ]].

At each SRV opening pressure, system pressure greater than rated is required to deliver rated system flow during steady-state operation. Therefore, the margins to the 125% mechanical overspeed trip for the HPCI and RCIC turbines are reduced. Additionally, the high vessel pressures have the potential to reduce the margin to the overspeed trips at the initial speed peak during the startup of the HPCI and RCIC systems. During a HPCI and RCIC start, the turbine governor valves are momentarily full open and therefore, the rate at which the speed increases is temporarily uncontrolled. Eventually, when hydraulic pressures enable the turbine control systems to take over the transient, the governor valve closes to control turbine speeds at the demanded flows. When a steady-state condition is reached, the final turbine speed is that indicated above, which is within the turbine speed limits of each system.

The potential concern during the startup transient is system availability. If the HPCI and RCIC turbines do trip during the startup, manual actions are required to reset the turbine trips. For HPCI, the turbine can be reset in the control room. For RCIC, the turbine must be reset locally.

The above considerations assume that HPCI and RCIC would initiate and operate when the reactor pressure is conservatively at the upper limit. The conclusions in the Low Low Set (LLS) discussion documented in Reference 5 remain unchanged as noted below. HPCI and RCIC will perform satisfactorily at a higher speed.

Regarding LLS valve operation, the increased SRV opening pressures will only affect the timing of the first SRV actuation. Once the logic is initiated, the opening and closing setpoints of pre-selected SRVs are automatically reset to lower values by the LLS logic. This logic is unaffected by the setpoint tolerance change because the logic acts on the relief mode of the SRV actuation and not on the safety mode of operation.

4.2 HPCI and RCIC Performance for Loss-of-Feedwater Events

For loss-of-feedwater events that do not isolate the reactor, vessel pressure is maintained by the turbine bypass valves at the Electro-Hydraulic Control (EHC) pressure setpoint. With vessel pressures at the EHC pressure setpoint which is lower than the SRV setpoints, HPCI and RCIC operation is not affected by an increase in SRV opening pressures. For MSIV closure events, the SRVs will actuate at the upper limit prior to the reactor water level reaching Level 2. The subsequent SRV actuations will be controlled by the LLS functions. Therefore, vessel pressures will be within the HPCI and RCIC design pressure range at the time of HPCI or RCIC initiation.

4.3 HPCI Performance for LOCA Events

The conclusions in the Low Low Set (LLS) discussion documented in Reference 5 remain unchanged as noted below. HPCI and RCIC will perform satisfactorily at a higher speed.

Regarding LLS valve operation, the increased SRV opening pressures will only affect the timing of the first SRV actuation. Once the logic is initiated, the opening and closing setpoints of pre-selected SRVs are automatically reset to lower values by the LLS logic. This logic is unaffected by the setpoint tolerance change because the logic acts on the relief mode of the SRV actuation and not on the safety mode of operation.

[[

]]

The discussions above demonstrate that SRV setpoint drift up to the upper limit has an insignificant effect on HPCI and RCIC performance.

5.0 Containment Evaluation

5.1 Objective and Scope

The purpose is to assess the effect of increasing HNP Safety Relief Valve (SRV) setpoint from 1150 psig to 1160 psig on the containment-related evaluations, which include the following.

1. Anticipated Transients Without Scram (ATWS)
2. Design Basis Accident (DBA) Loss-of-Coolant Accident (LOCA)
3. Small Steam Line Break (SSLB) for Equipment Qualification (EQ)
4. Appendix R
5. Station Blackout (SBO)

5.2 Design Inputs and Assumptions

The design inputs that are necessary to perform ATWS, DBA LOCA, SSLB for EQ, Appendix R and SBO are defined in Reference 2.

Consistent with Reference 7, ATWS analysis is separately performed for HNP Unit 1 and Unit 2 because the two units have different heat balance parameters (e.g., core flow, steam flow and feedwater temperature) and heat capacity temperature limit (HCTL) curves. The same inputs and assumptions as Reference 7 are applicable for SRV setpoint increase with the input clarification in Table 5-1. Consistent with Reference 7, the same Main Steam Isolation Valve Closure (MSIVC) event with the exposure of End of Cycle (EOC) is analyzed. The ODYN/STEMP analyses in Reference 6 provides the inputs for ATWS containment analysis based on SHEX.

The analyses for DBA LOCA and Appendix R in Reference 8 are also performed separately for HNP Units 1 and 2. The same inputs and assumptions as Reference 8 are applicable for SRV setpoint increase.

In Reference 8, SSLB for EQ and SBO analysis are performed based on the combined limiting input parameters for HNP Units 1 and 2. The same inputs and assumptions as Reference 8 are applicable for SRV setpoint increase.

5.3 Analysis Method

Consistent with Reference 7, the same ODYN, STEMP and SHEX methods are used for ATWS analysis. Consistent with Reference 8, the same SHEX method is used for SSLB for EQ analyses. Engineering evaluation is used for assessing the effects on DBA LOCA, Appendix R and SBO.

5.4 Analysis Results

5.4.1 ATWS

The containment responses for ATWS are summarized in Table 5-2.

[[

]] As shown in Table 5-2, there is no WW pressure change due to SRV setpoint increase to 1160 psig while the suppression pool temperatures are reduced by 1.3°F and 1.5°F for Unit 1 and Unit 2, respectively. Therefore, the existing ATWS containment analysis based on SHEX in Reference 7 remains applicable for SRV setpoint increase from 1150 psig to 1160 psig.

5.4.2 DBA LOCA

For DBA LOCA, [[

]].

Therefore, long-term DBA LOCA analysis in Reference 8 is not affected by SRV setpoint increase from 1150 to 1160 psig.

For short-term LOCA load, allowing for the 3% drift tolerance, the new setpoint (1160 psig + 3% = 1194.8 psig) is still lower than the Upper Limit value (1195 psig) that is used in the analysis in Reference 5. Therefore, SRV setpoint increase to 1160 psig has no effect on the short-term LOCA load in Reference 5.

5.4.3 SSLB for EQ

The peak dry well (DW) temperature for the SSLB EQ cases are summarized in Table 5-3. The peak values from Reference 8 are also included for purpose of comparison. As seen in the table, the changes on the peak values are insignificant (approximately -1°F). The changes on the DW temperature during the entire event are also insignificant (approximately +1°F/-1°F).

The DW temperature time histories and EQ envelope are plotted in Figure 5-1. The similar DW temperature responses as Reference 8 Figure D-1 are observed. Therefore, SRV setpoint increase from 1150 psig to 1160 psig has negligible effect on HNP Units 1 and 2 EQ profile.

5.4.4 Appendix R

It should be noted that an NRC safety evaluation was issued that transitioned the existing fire protection program (Appendix R) to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c).

5.4.4.1 RPV Inventory Response

Appendix R is a RPV isolation and non-break event, in which SRV is actuated and cycled after MSIVC occurs. Increasing the SRV setpoints to 1160 psig will [[

]] in comparison to the case with SRVs at the current setpoint of 1150 psig. However, the change [[

]] for the cases without spurious SRV operation (i.e., Cases 1 through 3 in the Hatch power uprate safety analysis report (PUSAR) in Reference 10) with the following reasons.

1) [[

]] by SRV setpoint increase to 1160 psig.

2) After first SRV actuation at 1160 psig, subsequent SRV actuations are on low-low-set logic, which remains unaffected by SRV setpoint increase to 1160 psig.

For Cases 4 and 5 in the Hatch PUSAR, the SRV setpoint increase to 1160 psig [[ ]] because of spurious SRV operation at event initiation (i.e., no SRV actuation at 1160 psig).

5.4.4.2 Containment Response

[[

]] As discussed in Section 5.4.4.1, [[ ]] for Cases 1 through 3 (Hatch PUSAR) without spurious SRV operation. Thus, the effect on the suppression pool temperature response due to SRV setpoint increase is negligible, and in turn, the effect on containment temperature and pressure are negligible.

As discussed in Section 5.4.4.1, the SRV setpoint increase [[ ]] for Cases 4 and 5 (Hatch PUSAR) because of spurious SRV operation at event initiation. Therefore, the containment response is not affected.

It is concluded that the Appendix R containment response in Reference 8 remains applicable for SRV setpoint increase to 1160 psig.

5.4.5 SBO

SBO is also a RPV isolation and non-break event similar to Appendix R. For SBO, there is no SRV spurious operation. After first actuation at 1160 psig, subsequent SRV actuations are on low-low-set logic to maintain vessel pressure until the end of 4 hour SBO coping period. The applicable discussion and conclusion for Appendix R in Section 5.4.4 are also applicable for SBO.

Table 5-1 ATWS Input Comparison

Parameters	ATWS - SHEX	ATWS - STEMP
Initial Suppression Pool Volume (ft^3)	86420(1)	85112 (U1)(2) 86420 (U2)(2) 86652 (U1)(3) 88045 (U2)(3)
Initial DW and WW Airspace Volume (ft^3)	262110 (U1) 259066 (U2)	262110 (U1) 259066 (U2)
Initial Condensate Storage Tank (CST) (lbm)(4)	4125000	3875755 (U1) 3471968 (U2)
Initial Condensate Storage Tank (CST) (ft^3)	66845(5)	62803 (U1) 56260 (U2)
Reference	7	9

(1) ATWS is a special event in which nominal assumptions can be used such as 1979 ANS 5.1 nominal decay heat that is used in Reference 7. Consistent with Reference 7, the minimum suppression pool volume for Unit 2 is used as nominal value for both units, which is still conservative (i.e., 86420 ft^3 is less than 86652 ft^3 (U1) and 88045 ft^3 (U2)).
(2) Minimum volume at low water level.
(3) Nominal volume that is based on average of maximum volume and minimum volume.
(4) Based on 14.7 psia and 120°F water temperature.
(5) For ATWS event, the CST inventory usage at the end of 4 hours is approximately 2%. Therefore, use of 66845 ft^3 in the analysis has no effect on the values that are reported in Table 5-2.

Table 5-2 ATWS Containment Results (MSIVC-EOC)

Parameter	HNP Unit 1		HNP Unit 2	
	Reference 7	SRV Setpoint	Reference 7	SRV Setpoint
Peak Wetwell Airspace Pressure (psig)	4.8	4.8	9.0	8.9
Peak Suppression Pool Temperature (°F)	213.6 @ 2875 sec	212.3 @ 2869 sec	215.3 @ 2911 sec	213.8 @ 2931 sec
Wetwell Pressure When Peak Suppression Pool Temperature Occurs (psig)	2.4	2.4	5.3	5.3

Table 5-3 SSLB Containment Results

Plant	Case	Peak DW Airspace Temperature (°F)	Time of Peak DW Airspace Temperature (sec)	Peak DW Shell Temperature (°F)	Time of Peak DW Shell Temperature (sec)
Reference 8					
HNP 1 & 2	0.01 ft^2 break	289	1800	255	1980
	0.10 ft^2 break	324	595	271	600
	0.50 ft^2 break	328	276	276	579
SRV Setpoint Increase					
HNP 1 & 2	0.01 ft^2 break	289	1770	255	1975
	0.10 ft^2 break	324	595	270	597
	0.50 ft^2 break	327	276	276	578

[[

]]

Figure 5-1 SSLB DW Temperature for EQ[1]

[1] For 0.5 ft^2 break, one case up to 1 day and one case up to 180 days are performed.

6.0 ATWS Mitigation Capability

6.1 Objective and Scope

The purpose of this evaluation is to assess the effect of increasing HNP Safety Relief Valve (SRV) Setpoint from 1150 psig to 1160 psig on HNP Anticipated Transients Without Scram (ATWS) transients. The assessment includes any potential effect of HNP SRV setpoint increase on key ATWS parameters in comparison with the corresponding ATWS acceptance criteria for limiting ATWS events.

6.2 Design Inputs and Assumptions

The design inputs that are necessary to perform the ATWS safety analysis are defined in the customer approved Design Input Request (DIR) (Reference 9). Because HNP Unit 1 and Unit 2 have unique heat balance parameters [[
]] as shown in Reference 9, the ATWS evaluations are performed based on a combination of operating conditions from Units 1 and 2 that are considered bounding for ATWS (unlike the ATWS containment results shown in Section 5.4 where there are separate analyses for both units). Therefore, the analysis results are applicable to Units 1 and 2.

The assumptions used in the HNP GNF3 New Fuel Introduction (NFI) ATWS analysis (Reference 11) based on assumptions allowed in ATWS analysis procedures if related are applicable to this analysis. No additional assumptions are made in this analysis.

6.3 Analysis Method

The limiting licensing basis ATWS events are analyzed to confirm the ATWS responses to the increase of SRV opening setpoint from 1150 psig to 1160 psig meet the corresponding ATWS acceptance criteria listed below. The limiting ATWS events of Main Steam Isolation Valve Closure (MSIVC) and Pressure Regulator Failure Open – Maximum Steam Demand (PRFO) are analyzed at Beginning of Cycle (BOC) and End of Cycle (EOC) conditions.

The limiting ATWS events are evaluated at rated power to demonstrate compliance to the following.

1. ASME Service Level C Pressure Limit (1500 psig)
2. Containment Pressure Design Limit [[ ]]
3. Suppression Pool Temperature Design Limit [[
]]
4. 10CFR50.46 Peak Cladding Temperature (PCT) Limit (<2200°F)
5. 10CFR50.46 Fuel Local Cladding Oxidation Thickness Limit (<17%)

The effects on peak vessel pressure, peak suppression pool temperature, and containment pressure are explicitly analyzed in the ATWS analysis. The PCT and fuel local cladding oxidation are justified for compliance with the corresponding acceptance criteria based on the large margins and historical PCT results for other plants as discussed in Reference 4.

GNF3 fuel design cycle-independent analyses show that an ODYN peak HNP NPSH suppression pool temperature limit of 217.0°F for both Units will ensure that the SHEX results of Reference 7 remain valid.

6.4 Analysis Results

6.4.1 Vessel Pressure

[[

]]

6.4.2 Suppression Pool Temperature and Containment Pressure

[[

]] Furthermore, large margins exist relative to the suppression pool temperature and containment pressure design limits. See Table 6-2 for associated margins.

6.4.3 PCT and Cladding Oxidation

The increase in SRV setpoints has a no effect on the peak cladding temperature result. [[

]] Furthermore, significant margin exists relative to the PCT limit per Reference 4. There are no cladding oxidation thickness concerns [[ ]]. Therefore, the PCT and local cladding oxidation thickness acceptance criteria are still met for the increase of SRV opening setpoint from 1150 psig to 1160 psig for HNP Units 1 and 2.

6.4.4 Additional ATWS Results

Table 6-1 provides an additional summary of detailed ATWS results.

6.4.5 Summary of Results

The limiting ATWS analysis results in comparison with the corresponding acceptance criteria are shown in Table 6-2. Based on the analysis results, all ATWS acceptance criteria are met for the increase of SRV opening setpoint from 1150 psig to 1160 psig for HNP Units 1 and 2. The ATWS analysis results are applicable to mixed cores of GNF2 and GNF3, as well as full cores of GNF3.

Therefore, the increase of HNP SRV setpoint from 1150 psig to 1160 psig is acceptable regarding ATWS acceptance criteria compliance.

The limiting peak pressure [[ ]] is 1218 psia from MSIVC at BOC case with SLCS initiation time of 130.6 sec. These results are provided to support further assessment of the SLCS discharge pressure.

Table 6-1 Summary of Additional ATWS Analysis Results

Initiating Event	Exposure	Peak Neutron Flux (%)	Peak Heat Flux (%)	Peak Dome Pressure (psig)	Peak RPV Pressure (psig)	Peak Pool Temperature (°F)
PRFO	BOC	350	159	1438	[[	
	EOC	426	164	1410		
MSIVC	BOC	252	140	1412		
	EOC	300	146	1402		]]

Table 6-2 ATWS Analysis Results and Criteria

Item	Parameter	Unit	Result[1]	GNF3 NFI Result[2]	Limit	Acceptance Criteria Met?
1	Peak Vessel Bottom Pressure	psig	[[		1500	Yes
2	Peak Suppression Pool Temperature	°F			217	Yes
3	Peak Containment Pressure	psig			56	Yes
4	Peak Cladding Temperature	°F			2200	Yes
5	Cladding Oxidation Thickness	%		]]	17	Yes

1. [[ ]]
2. [[ ]]

7.0 Conclusions

As was noted above, the following evaluations were performed to address the increase of the SRV opening setpoints.

- Loss of Coolant Accident (LOCA)
- High Pressure System Performance
- Containment Performance
- Anticipated Transients Without Scram (ATWS)

All evaluations have shown that the increased setpoint of 1160 psig yield adequate performance results. Given this information, HNP can use this document to support a license amendment request for increasing the nominal setpoint.

It should be noted that Southern Nuclear Operating Company should verify compliance with NRC Generic Letter 89-10 (Reference 3) requirements for valves in HPCI and RCIC.

8.0 References

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2. SNC Letter NMP-ES-050-F01, RER Number: SNC1512291 Sequence No.: 2, Letter from David Sanford (SNC) to Jarrod Miller (GNF-A), "Hatch SRV Setpoint Increase - DBR-0075058 and DBR-0075139," July 28, 2023.
3. NRC GL 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance," June 28, 1989.
4. NEDC-33879P, Revision 2, "GNF3 Generic Compliance with NEDE-24011-P-A (GESTAR II)," March 2018.
5. NEDC-32041P, Revision 2, GE Nuclear Energy, "Safety Review for Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Updated Safety Relief Valve Performance Requirements," April 1996.
6. GEH Report 008N0745, Revision 0, "Hatch Nuclear Plant SRV Setpoint Increase – ATWS Analysis," November 2023.
7. GEH Report 0000-0106-1182, Revision 0, "Edwin I. Hatch Units 1 and 2 Ultimate Heat Sink Temperature Increase to 97°F Impact on Anticipated Transients Without Scram (ATWS) Event Containment Analysis," August 2011.
8. GEH Report 004N8577, Revision 0, "Edwin I. Hatch Nuclear Power Plant Units 1 and 2 Containment Analyses for GNF3 New Fuel Introduction," November 2018.
9. SNC Letter NMP-ES-050-F01, Letter from David Sanford (SNC) to Jarrod Miller (GEH), "Hatch SRV Setpoint Increase – DBR-0075302," August 7, 2023.
10. NEDC-32749P, Revision 0, "Extended Power Uprate Safety Analysis Report for E.I. Hatch Plant Units 1 and 2," July 1997.
11. 004N6886, Revision 0, "GNF3 Fuel Design Cycle-Independent Analyses for Edwin I. Hatch Nuclear Plant Units 1 and 2," November 2018.