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10 CFR 50.90

JAFP-24-0010

February 29, 2024

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

Subject: Response to Request for Additional Information for License Amendment Request to Modify Technical Specification Surveillance Requirement (SR) 3.4.3.1 Safety Relief Valves (S/RVs) Setpoint Lower Tolerance

References:

1. Letter from D. Gudger (Constellation Energy Generation, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request to Modify Technical Specification Surveillance Requirement (SR) 3.4.3.1 Safety Relief Valves (S/RVs) Setpoint Lower Tolerance," JAFP-23-0038, dated July 28, 2023
2. Email from James Kim (Project Manager, U.S. Nuclear Regulatory Commission) to A Hasanat (Constellation Energy Generation, LLC), "FitzPatrick - Final SNSB RAI regarding Amendment to Modify Safety Relief Valves Setpoint Lower Tolerance (EPID: L-2023-LLA-0103)," ML24024A137, dated January 24, 2024

By letter dated July 28, 2023 (Reference 1), Constellation Energy Generation, LLC (CEG) requested to change the James A. FitzPatrick Nuclear Power Plant (JAF) Technical Specifications (TS). The proposed amendment request is to modify Technical Specification Surveillance Requirement (SR) 3.4.3.1 Safety Relief Valves (S/RVs) Setpoint Lower Tolerance.

On January 24, 2024 (Reference 2), the U.S. Nuclear Regulatory Commission (NRC) determined that additional information was necessary to complete the review of the amendment.

Attachment 1 to this letter contains the NRC's request for additional information immediately followed by CEG's response.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards considerations.

This amendment request contains no new regulatory commitments.

Should you have any questions concerning this submittal, please contact Abul Hasanat Abul.Hasanat@Constellation.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29th day February 2024.

Respectfully,

David T. Gudger

David T. Gudger
Senior Manager - Licensing & Regulatory Affairs
Constellation Energy Generation, LLC

Attachment 1: Response to Request for Additional Information

cc:	USNRC Region I, Regional Administrator	w/attachments
	USNRC Senior Resident Inspector, JAF	"
	USNRC Project Manager, JAF	"
	A. L. Peterson, NYSERDA	"
	B. Frymire, NYSPSC	"
	C. Powers, NYSPSC	"

ATTACHMENT 1

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

Response to Request for Additional Information

INTRODUCTION

By application dated July 28, 2023, Reference 1, Constellation Energy Generation, LLC (CEG) submitted a license amendment request (LAR) requesting change to the Technical Specifications (TS) for the James A. FitzPatrick Nuclear Power Plant (JAF) Renewed Facility Operating License DPR-59. The proposed change would revise the safety function lift setpoint tolerances for the Safety/Relief Valves (S/RVs) that are listed in Surveillance Requirement (SR) 3.4.3.1 of the Technical Specifications (TS). This change would be limited to the lower tolerances and would not affect the upper limits. The tolerance band for these valves would be changed from ±3% to +3% or -5% of the setpoint (1145 psig +34.3 or -57.2 psig).

After reviewing the LAR, (Reference 1), the NRC staff requested the additional information (RAI) given below, excerpted from Reference 3.

SNSB-RAI 1

Regulatory Basis:

10 CFR 50, Appendix A, GDC Criterion 16, "Containment design." Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

10 CFR 50, Appendix A, GDC Criterion 50, "Containment design basis." The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

RAI:

Refer to General Electric (GE) proprietary report NEDE-22223, "Low-Low Set Logic and Lower MSIV Water Level Trip for BWRs with Mark I Containment," dated September 1982 (Reference 2) which proposed design modifications for BWR/2-4, with Mark I containments to limit containment loading following subsequent S/RV actuations. The intent of these modifications was to reduce the discharge loads on the containment and suppression pool structures resulting from subsequent S/RV actuations during a transient.

The proposed modifications consisted of a low-low set (LLS) relief logic system, and a lower main steam isolation valve (MSIV) water level trip. The LLS is an automatic S/RV control system while the lower MSIV water level trip modification is applicable to BWR/4 plants only. As JAF is a BWR/4, Mark I containment plant:

(a) Did JAF implement the recommended modifications for BWR/4 with Mark I containments as described in NEDE-22223?

(b) The modifications (if implemented) were developed with the lower setpoint at -3% of the value. The submitted LAR proposes a change to the lower setpoint for the S/RVs from -3% to -5%. Please describe the impact this setpoint change has on containment loads during subsequent S/RV actuations during transients as reported in NEDE-22223.

(c) If JAF has not implemented the recommended modifications, provide reasons for not implementing the recommended changes and describe the impact the proposed setpoint tolerance change has on containment loading during subsequent S/RV actuations during the applicable transients.

Response to SNSB-RAI 1:

(a) JAF partially implemented the recommended modifications for BWR/4 with Mark I containments as described in NEDE-22223. Changes to MSIV Low Water Level Setpoint consistent with the recommendations in NEDE-22223 were made in Technical Specification Amendment (TSA) 103, Reference 4.

(b) JAF did not implement the LLS Relief Logic.

In 1983, JAF performed plant unique analysis of S/RV transient response which was prepared accordance with the MARK I Containment Long Term Program, NUREG-0661 (JPN-83-088) in Reference 5. As the LLS modification was not performed, there is no S/RV opening logic associated with the low allowable value of the safety relief valve mechanical setpoints. Current dynamic load analysis, conservatively use an upper bound of the allowable value (1145 psig + 3%) as the assumed S/RV lift pressure. An earlier S/RV lifting pressure would result in reduced containment and S/RV discharge piping transient loading. The transient evaluations consider the second actuation to occur at the maximum reflood and demonstrated that allowable stresses are not exceeded.

The lowering of the allowable value is bounded by existing analysis and does not present an adverse change to containment loads.

(c) JAF did not implement all recommended modifications of NEDE-22223. As discussed above, JAF met Mark I Containment Long Term Program stress requirements absent the implementation of LLS by meeting stress allowable criteria for subsequent actuations at the most limiting timing.

During the review of NEDE-22223, it was identified that BWR Owners group provided recommendations for proposed actions and modifications to meet the objectives of NUREG-0737, Item II.K.3.16. These actions and modifications were designed to reduce subsequent actuations for plant transients, reactor isolations, and improve overall S/RV performance. The recommendations included:

1) LLS Relief Logic System or Equivalent Manual Actions

2) Lower the reactor pressure vessel water level isolation setpoint for main steam isolation valve closure from Level 2 to Level 1

JAF's response was that equivalent operator manual actions as implemented by emergency procedures adequately reduced the challenges and failures of relief valves in coordination with preventative maintenance practices and achieves the objectives of NUREG-0737, Item II.K.3.16.

These equivalent manual actions remain in place. Current emergency operating procedures for high reactor pressure (greater than 1080 psig) prompt entry into emergency operating procedures for Hot RPV Control, which directs operators to stabilize pressure below 1080 psig, which is well below the lifting setpoint for S/RV actuation.

Subsequent to this response, JAF implemented TSA 103 (Reference 4) to lower the MSIV setpoint to Level 1 as recommended.

The proposed change to the allowable value is bounded by the existing analyses.

SNSB-RAI 2

Regulatory Basis:

10 CFR 50, Appendix A GDC Criterion 13, "Instrumentation and control." Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

10 CFR 50, Appendix A, GDC 29, "Protection against anticipated operational occurrences." The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

RAI:

Reference 1 discusses operating margin in relation to nominal operating pressure in Section 3.3 of Attachment 1, with the statement:

"The purpose of the lower setpoint tolerance is to ensure sufficient margin exists between the normal operating pressure of the system and the point at which the S/RVs actuate in the overpressure safety mode. The nominal operating pressure of the reactor pressure vessel power is 1040 psig. A lower setpoint tolerance value of -5%, applied to the S/RV set pressure (1145 psig) would allow it to lift at 1087.8 psig."

(a) What is the tolerance (uncertainty) range of the reactor instrumentation that provides a nominal reading of 1040 psig?

(b) Please describe the impact of the proposed change and justify sufficient margin remains between the upper tolerance limit of the normal nominal operating pressure and the proposed lowest S/RV opening pressure of 1087.8 psig so that the S/RVs would not actuate during normal plant operation.

    Similarly, in TS Table 3.3.1.1-1, the reactor protection system (RPS) Function 3 has the reactor pressure allowable value ≤ 1080 psig.

(c) Provide the RPS scram setpoint during an overpressure transient and the margin between the scram setpoint pressure and the proposed lowest S/RV opening pressure of 1087.8 psig. Justify the margin is sufficient so that the S/RV would not operate prior to the RPS actuation.

Response to SNSB-RAI 2:

(a) Narrow range reactor pressure indication is provided by the Feedwater Narrow Range Pressure Control Transmitter and associated recorder. This instrument has a total loop uncertainty of 7.55 psi.

Among other indications of reactor pressure in the control room, the feedwater level control system provides other indications of reactor pressure using the wide range reactor pressure indication. The wide range reactor pressure indication is associated with the high reactor pressure alarm at a setpoint of 1051 psig and draws from inputs from the Reactor Water Level Feedwater Control Compensating Pressure Transmitters. This recorder and associated alarm have a total loop uncertainty of 19.2 psi.

(b) With reactor pressure instrumentation available in the control room, the margin to detect deviation from nominal reactor pressure at 1040 psig is based on the total loop uncertainty of Feedwater Narrow Range Pressure Control Transmitter at 7.55 psig. Reactor pressure is set during OP-65, Startup and Shutdown Procedure, at between 1031 and 1040 psig. This creates a maximum deviation up to 1047.55 psig, which is 40.25 psi of margin. Deviations in reactor pressure during operation would be detected during daily surveillances of reactor pressure instrumentation. With an alarm setpoint of 1051 psig and a total loop uncertainty of 19.2 psi, the reactor high pressure alarm is ensured to actuate prior to 1070.2 psig. This is 17.6 psi of margin to the proposed low allowable value for the mechanical lift.

(c) Normal operation is ultimately bounded by reactor pressure indication for the RPS High Pressure Scram, with a Technical Specifications Allowable Value of 1080 psig. The lowest proposed allowable value associated with this amendment for an S/RV mechanical lift is 1087.8 psig. This amendment does not propose revising the +/- 1% as-left values or nominal setpoints.

The RPS high pressure scram setpoint is established at 1062 psig, to ensure that the allowable value of 1080 psig is not exceeded. The margin between the scram setpoint pressure and the proposed lowest allowable S/RV opening pressure is 25.8 psi. This margin is justified based on the total loop uncertainty of 14.25 psi.

For steady state operation or gradual rises in reactor pressure, the spurious opening of S/RVs which are within the proposed allowable ranges is ensured by the reactor protection system as described above. Adequate margin exists from normal reactor instrumentation and Reactor Protection System high pressure allowable values to prevent the lower allowable values of the SRV mechanical lift from impacting normal operation.

Overpressure transients are evaluated in the design basis abnormal operational transients in section 14.5.2 of the Updated Final Safety Analysis Report (UFSAR).

These transients include:

a. Generator Trip (turbine control valve fast closure)
b. Turbine Trip (turbine stop valve closure)
c. Closure of the main steam line isolation valves
d. Turbine pressure regulator failure to minimum demand
e. Failure of the Turbine Bypass Valves to open when required
f. Loss of main condenser vacuum

These transients assume that reactor pressure will rise and result in the automatic opening of S/RVs. The reactor scram may be initiated by the high reactor pressure or high reactor flux setpoints. S/RVs are subsequently assumed to operate to control reactor pressure. These analyses do not use the lower allowable value as an input.

The loss of feedwater abnormal operational transient was additionally reviewed, and S/RVs are assumed to not operate due to the lowering of MSIV water level isolation to Level 1 and reactor pressure being controlled by the main turbine bypass valves.

The operation of S/RVs during overpressure transients at a lower mechanical lift allowable value is bounded by existing design basis analyses of abnormal operational transients.

REFERENCES

1. Letter from CEG to NRC, License Amendment Request to Modify Technical Specification Surveillance Requirement (SR) 3.4.3.1 Safety Relief Valves (S/RVs) Setpoint Lower Tolerance, ML23209A003, July 28, 2023
2. NEDE-22223, Low-Low Set Logic and Lower MSIV Water Level Trip for BWRs with Mark I Containment, ML19262G927, dated September 1982
3. NRC email, FitzPatrick - Final SNSB RAI regarding Amendment to Modify Safety Relief Valves Setpoint Lower Tolerance (EPID: L-2023-LLA-0103), ML24024A137, dated January 24, 2024
4. NRC letter, Safety Evaluation Supporting Amendment No. 103 to Facility Operating License No. DPR-59 Power Authority of the State of New York James A. FitzPatrick Nuclear Power Plant Docket No. 50-333, ML010610096, dated December 19, 1986
5. FitzPatrick letter, Mark I Containment Program Plant Unique Analysis Report, ML20078H509, JPN-83-88, dated October 11, 1983