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10 CFR 50.90

JAFP-23-0038

July 28, 2023

U.S. 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: License Amendment Request to Modify Technical Specification Surveillance Requirement (SR) 3.4.3.1 Safety Relief Valves (S/RVs) Setpoint Lower Tolerance

In accordance with the provisions of Title 10 of the Code of Federal Regulations with the provisions of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Constellation Energy Generation, LLC (CEG) hereby requests an amendment to Renewed Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant (JAF). The proposed amendment would expand 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 (TSs). 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.

The as-left tolerance band for the safety function lift setpoints will continue to be ±1%, but the as-found tolerance band will change to +3% or -5%. If a valve is tested and the lift setpoint is found outside the tolerances, the valve failure will be evaluated in the JAF Corrective Action Program. The as-found tolerance band is used for determining operability and to increase sample size for Inservice Testing (IST).

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. The bases for these determinations are included in Attachment 1 of this submittal.

The proposed TS markup pages are included as Attachment 2 to this submittal. Markups of the proposed TS Bases are included for information only as Attachment 3 of this submittal. Clean pages of the proposed TS changes are included as Attachment 4 of this submittal.

This amendment request contains no new regulatory commitments.

CEG requests approval of the proposed amendment by July 31, 2024. Once approved, the amendment shall be implemented within 45 days.

The proposed changes have been reviewed by the JAF Plant Operations Review Committee in accordance with the requirements of the Constellation Quality Assurance Program.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), CEG is transmitting a copy of this application and its attachments to the designated State Officials.

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 28th day of July 2023.

Respectfully,

David T. Gudger

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

Attachments:

1. Evaluation of Proposed Changes
2. Proposed Technical Specification Marked-Up Pages
3. Proposed Technical Specification Bases Marked-Up Pages
4. Proposed Technical Specification Clean Pages

cc:	USNRC Region I, Regional Administrator	w/attachments
	USNRC Senior Resident Inspector, JAF	w/attachments
	USNRC Project Manager, JAF	w/attachments
	A. L. Peterson, NYSERDA	w/attachments

ATTACHMENT 1

License Amendment Request

James A. FitzPatrick Nuclear Power Plant

Docket No. 50-333

EVALUATION OF PROPOSED CHANGES

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

1.0 SUMMARY DESCRIPTION

Constellation Energy Generation, LLC (CEG) proposes to revise the FitzPatrick Nuclear Power Plant (JAF) Technical Specifications (TS) 3.4, "Reactor Coolant System (RCS)," Section 3.4.3, "Safety/Relief Valves (S/RVs)." Specifically, CEG proposes a new safety function lift setpoint lower tolerance for the S/RVs as delineated in SR 3.4.3.1. The proposed change will revise the lower setpoint tolerance from -3 percent (%) to -5%.

This change is limited to the lower tolerances and does not affect the upper tolerances; the upper tolerance will remain at +3% of the safety function lift setpoint. In addition, this change only applies to the as-found tolerance and not to the as-left tolerance, which will remain unchanged at ±1 % of the safety lift setpoint. The as-found tolerances are used for determining operability and to increase sample sizes for S/RV testing should the tolerance be exceeded. There will be no revision to the actual setpoint of the valves installed in the plant due to this change.

The proposed change relaxes an unnecessarily restrictive Surveillance Requirements (SR). The proposed change will not impact the reliability of the S/RVs or adversely impact their ability to perform their safety function. This change will preclude the submittal of previously reportable Licensee Event Reports (LERs) to the U.S. Nuclear Regulatory Commission (NRC) due to setpoint drift in the low (conservative) direction.

2.0 DETAILED DESCRIPTION

The proposed amendment would change the safety function lift setpoint lower tolerance for the S/RVs that are listed in TS:

- 3.4.3, Safety/Relief Valves (S/RVs)

The tolerance band for S/RVs would be changed from ±3**%** to +3 or -5% of the safety lift function setpoint. This change only applies to the as-found tolerance band and not to the as-left tolerance band which will remain at ±1% of the safety function lift setpoint. The as-found tolerances are used for determining operability and to increase sample sizes for testing. There will be no change to the valves as installed in the plant. The following are the current and proposed setpoint tolerance values which apply to SR 3.4.3.1:

Number of S/RVs	Current Setpoint (psig)	Proposed Setpoint (psig)
11	1145 ±34.3	1145 +34.3 or -57.2

Note: the setpoint tolerance values have been rounded in a conservative direction.

The proposed change relaxes an unnecessarily restrictive surveillance requirement. The proposed change will not impact the reliability of the S/RVs or adversely impact their ability to perform their safety function. The S/RVs are required to meet the American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code limits based on valve type and size to ensure acceptable valve performance (Reference 6.5). These limits are not being changed. This change will preclude the submittal of certain LERs to the NRC due to setpoint drift in the low (conservative) direction.

3.0 TECHNICAL EVALUATION

3.1 Background

At JAF, the Reactor Coolant System Pressure Relief System consists of S/RVs located on the main steam (MS) lines between the reactor vessel and the first isolation valve within the drywell. These valves protect against overpressure of the reactor coolant system.

The S/RVs provide three main protection functions:

a. Overpressure relief operation - The valves open automatically to limit pressure rise.

b. Overpressure safety operation - The valves function as safety valves and open (self-actuated operation if not automatically opened for relief operation) to prevent nuclear system over pressurization.

c. Depressurization operation - The automatic depressurization system (ADS) valves open automatically as part of the emergency core cooling system (ECCS) for events involving small breaks in the nuclear system process barrier (reactor coolant pressure boundary).

In the safety mode, or the spring mode of operation, the valves open when steam pressure at the valve inlet overcomes the spring force holding the valve closed. This mode satisfies the ASME Code requirements. It is this mode of operation for which the lower surveillance tolerances for the safety function as-found lift setpoints will be relaxed from -3% to -5%. The as-found upper surveillance tolerances will remain at +3%. The relief and automatic depressurization modes rely upon solenoid actuation to open the valves and are not affected by this proposed change.

The JAF S/RVs are Target Rock three stage safety/relief valves, model 0867F-001 originally built to ASME Section III, 1968 editions and addenda through Summer, 1970 (Reference 6.4).

A review of as-found test data for the JAF S/RVs indicates a tendency for minor setpoint drift in the negative direction. JAF experience shows that it is the nature of these valves to have a drift/variance with an initial as-found low lift pressure.

Currently, a partial compliment of S/RVs are removed during each refueling outage, bench tested for safety set pressure and replaced with valves certified to have zero seat-to-disk leakage and to have safety lift setpoint tolerances within ±1% of the setpoint as specified in the TS SR 3.4.3.1. If the as-found lift is greater than the ±3% tolerance for one of the S/RVs tested from the original sample size, the sample size will be increased by two S/RVs, in accordance with the Inservice Testing (IST) Program requirements.

On March 8, 1993, the NRC issued a Safety Evaluation (SE) (Reference 6.2) for the GE Nuclear Energy Licensing Topical Report (LTR) NEDC-31753P (Reference 6.1) submitted by the Boiling Water Reactor Owners Group (BWROG). In the SE, the NRC stated that a generic change of setpoint tolerance to ±3% is acceptable provided that it is evaluated in the analytical bases. The required analysis was completed for JAF and the change was approved by the NRC (Reference 6.3).

The operability of the S/RVs is based on the TS SR acceptance criteria with a setpoint tolerance of ±3%. If any S/RV exceeds the tolerance, an Issue Report (IR) for each S/RV that exceeds the tolerance is entered into the JAF Corrective Action Program to evaluate the test failure. In addition, test failures outside of ±3% would result in testing additional valves to comply with the ASME OM Code requirements.

3.2 Evaluation

The proposed lower setpoint tolerance change from -3% to -5%, was evaluated using the previously accepted methodology of the LTR and the associated SER. Since the evaluation performed in detail to support the current upper and lower tolerances of ±3%, and the conclusions of the evaluation have not changed, only those areas not previously reviewed by the NRC are included in this evaluation.

3.2.1 Thermal Limits

The effect of adjusting the lower S/RV tolerance from -3% to -5% on Anticipated Operational Occurrences has been evaluated for the Operating Limit Minimum Critical Power Ratios (MCPRs), off-rated MCPRs, and off-rated Linear Heat Generation Rates (LHGRs). The thermal limits Anticipated Operational Occurrence (AOO) events are not affected because:

- Thermal limits AOOs credit the relief mode opening of the S/RVs. The Reload Licensing Analysis setpoints used for transient analysis are greater than the allowable Tech Spec +3% upper tolerance, and are not affected by an increase to the lower tolerance.

- S/RV opening (due to either relief mode actuation or safety mode actuation) typically occurs after the time of thermal limits minimums. Therefore, the resulting Operating Limit MCPR and adherence to Thermal Overpower and Mechanical Overpower limits are not significantly impacted by the opening of S/RVs.

- If S/RVs do open prior to the time of thermal limits minimums, later opening of the valves provides more conservative results. The pressure feedback on the core would be larger with later opening times such that the power feedback would be more severe, resulting in a worse thermal limits response. Thus, opening the valves earlier would only result in a thermal limits improvement.

For the above reasons, the expansion of the spring safety valve setpoint uncertainty from -3% to -5% is negligible to thermal limits AOO analyses. The thermal limits event analyses and the resulting MCPR and LHGR limits remain valid and unchanged.

3.2.2 ASME Overpressure

A postulated Main Steam Isolation Valve (MSIV) fast closure is the limiting event for peak vessel pressurization for ASME overpressure consideration, where the vessel dome (and thus bottom) pressure is maximized. In any pressurization case, an earlier opening of the S/RVs due to lower setpoints would produce improved results and gain margin. Analysis of the ASME Overpressure peak pressure event is conservatively based on the S/RV upper tolerance settings (i.e., +3% setting). Therefore, lowering the S/RV tolerance range from -3% to -5% does not have an adverse impact on the ASME overpressure analysis result.

3.2.3 LOCA Analysis

The Loss of Coolant Accident (LOCA) credits the ADS function of the S/RVs during a small break. The ADS function is not based on pressure setpoints. During a large break LOCA the reactor vessel is automatically depressurized due to the nature of the event and the S/RV safety function is not actuated (Reference 6.4). Therefore, in any type of LOCA there is no adverse impact from lowering the S/RV tolerance range from -3% to -5%.

3.2.4 High Pressure System Performance

The high pressure systems include Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) as well as Standby Liquid Control (SLC) system. The most significant effect of changing the S/RV setpoint tolerance on the HPCI and RCIC and SLC systems operations is the maximum reactor pressure at which they are required to deliver flow to the reactor. Since the limiting safety analyses are conservatively based on the S/RV upper tolerance settings (i.e., +3% setting), lowering the S/RV tolerance range from -3% to -5% does not have an adverse impact on the high pressure system performance.

3.2.5 ATWS Mitigation (Peak Pressure Suppression Pool Temperature)

The limiting Anticipated Transient Without Scram (ATWS) case for peak reactor pressure is similar to the ASME Overpressure event (MSIV closure) discussed above, except that a complete failure of scram is assumed. In any pressurization case, an earlier opening of the S/RVs due to lower setpoints would produce improved results and gain margin. Analysis of the ATWS peak pressure and suppression pool temperature are conservatively based on the S/RV upper tolerance settings (i.e., +3% setting). Therefore, lowering the S/RV tolerance range from -3% to -5% does not have an adverse impact on the ATWS overpressure or suppression pool temperature analysis results.

3.2.6 Containment Analyses

Containment analyses such as peak suppression pool temperature, peak suppression pool pressure, discharge line dynamic load, and submerged structure loads are all most limited by high mass/energy flow rates from the vessel into the containment structure. The flow rate through the S/RVs is directly proportional to the pressure differential across them. Since allowing lower pressure differences at the minimum end does not affect the limiting analyses at the maximum end of the pressure range, there is no impact on the containment systems, structures, or components' ability to manage energy release during S/RV actuation. Current containment analyses bound the proposed change.

3.3 Operating Margin

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. The lowest potential margin between the nominal reactor pressure and the S/RV lift pressure of the valve’s safety setpoint is 47.8 psig. The lowest potential margin between the nominal reactor pressure and the S/RV lift pressure of the valve's safety setpoint is changing from 70.7 psig to 47.8 psig.

The requested SR 3.4.3.1 revision will lower the minimum S/RV as-found set point acceptance criteria from 1110.7 psig to 1087.8 psig.

The proposed revision to SR 3.4.3.1 only changes the S/RV setpoint acceptance criteria minimum tolerance. The as-left or as-installed setpoint tolerances are unchanged; therefore, the revision does not affect the actual operation of the S/RVs. The requested SR 3.4.3.1 revision has no effect on the S/RV design basis such as simmering, seat leakage, or valve reliability.

The valves removed for testing are returned to a tolerance of ±1% prior to being installed for service, thereby returning the margin to the original levels. Therefore, the margin is considered adequate and will not impact normal plant operation.

3.4 Surveillance Test History

Test results for JAF S/RVs have shown that approximately 31% of S/RVs experience minor setpoint drift sufficient to exceed the acceptance criteria of -3% in the negative direction over time. Based on a review of the 16 previous test results shown in Table 1 for the JAF valves the average drift was -2.28%. There were 5 valve tests that failed below the -3% tolerance with none of these valves testing below -5%. The data includes tests for valves that span up to 4 years between bench tests. Therefore, excessive drift over six years is not anticipated for the S/RVs.

Table 1

S/RV Equipment No.	Year/Outage	Test Type	S/RV S/N	Set Pressure	Tolerance	As-found	Deviation
71A	2020 / J1R24	C	91	1145	±3%	1112	-2.9%
71C	2020 / J1R24	C	51	1145	±3%	1130	-1.3%
71E	2020 / J1R24	C	52	1145	±3%	1168	2.0%
71J	2020 / J1R24	C	48	1145	±3%	1120	-2.2%
71K	2020 / J1R24	C	95	1145	±3%	1128	-1.5%
71B	2022 / J1R25	C	90	1145	±3%	1094	-4.5%
71D	2022 / J1R25	C	97	1145	±3%	1105	-3.5%
71F	2022 / J1R25	C	49	1145	±3%	1134	-0.9%
71G	2022 / J1R25	C	27	1145	±3%	1100	-3.9%
71H	2022 / J1R25	C	53	1145	±3%	1122	-2.1%
71L	2022 / J1R25	C	85	1145	±3%	1131	-1.2%
71A	2022 / J1R25	CA	119	1145	±3%	1109	-3.1%
71C	2022 / J1R25	CA	25	1145	±3%	1118	-2.4%
71E	2022 / J1R25	CA	22	1145	±3%	1113	-2.9%
71J	2022 / J1R25	CA	21	1145	±3%	1099	-4.0%
71K	2022 / J1R25	CA	24	1145	±3%	1121	-2.1%

Table Key

S/N Pilot Serial Number

Test Type

C Test for ASME Code and TS Compliance

CA Additional S/RV Tested due to ASME Code Test Failure

4.0 REGULATORY EVALUATION

JAF Updated Final Safety Analysis Report (UFSAR) (Reference 6.4) section 16.6, "Conformance to AEC Design Criteria," provide detailed discussion of JAF's compliance with the applicable regulatory requirements and guidance.

The proposed TS amendment:

a. Do not alter the design or function of any system;

b. Do not result in any change in the qualifications of any component; and

c. Do not result in the reclassification of any component's status in the areas of shared, safety-related, independent, redundant, and physically or electrically separated.

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 Applicable 10 CFR 50 Appendix A General Design Criteria (GDC)

The following GDCs for the RCS, which require that the system be protected from overpressurization, were evaluated to determine if these GDC continue to be met.

- GDC 15 - Reactor Coolant System Design (Criterion 15)

  The reactor coolant system (RCS) and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).

  An example of the integrated protective action scheme, which provides sufficient margin to ensure that the design conditions of the RCPB are not exceeded, is the automatic initiation of the nuclear system pressure relief system upon receipt of an overpressure signal. To accomplish overpressure protection, a number of pressure-operated relief valves are provided to discharge steam from the nuclear system to the suppression pool. The nuclear system pressure relief system also provides for automatic depressurization of the nuclear system in the event of a LOCA in which the vessel is not depressurized by the accident. The depressurization of the nuclear system in this situation allows operation of the low pressure ECCS to supply enough cooling water to adequately cool the core. In a similar manner, other auxiliary, control, and protection systems ensure that the design conditions of the RCPB are not exceeded during any conditions of normal operation, including AOOs. Criterion 15 is not affected by this proposed change.

- GDC 35 - Emergency Core Cooling (Criterion 35)

  A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued *effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.*

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The ADS functions to reduce the reactor pressure so that flow from Low Pressure Coolant Injection (LPCI) and enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the nuclear system pressure relief valves to relieve the high pressure steam to the suppression pool. Criterion 35 is not affected by this proposed change.

Thus, the proposed change to the S/RV lower setpoint tolerances does not change the conformance with the above GDC and these GDC will continue to be met by this LAR.

4.1.2 Applicable ASME Code Requirements

The JAF IST program is currently implemented in accordance with the requirements of the ASME Operation and Maintenance (OM) Code 2004 Edition through OMb-2006 Addenda. The S/RVs at JAF are Class 1 Category C valves in accordance with the JAF IST Program. As required by the ASME OM Code, additional valves would be tested if the as-found setpoint of a tested valve from the sample exceeds ±3% of the nameplate set pressure. If one of these additional valves fails, then all of the main steam S/RVs would be removed and tested. OM Code testing of all S/RVs is accomplished over a period of five years. The JAF S/RVs will continue to be tested in accordance with these requirements.

The ASME OM Code Mandatory Appendix I, Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants, Guiding Principles, paragraph I-1310(e), Acceptance Criteria, allows the owner to establish setpoint acceptance criteria for relief valves tested under the IST Program; therefore, no relief will be required with regard to the setpoint tolerance change from -3% to -5%. However, a change to the JAF TS will be required and is being proposed with this LAR.

4.2 Precedent

Additional nuclear power generating stations have received NRC approval to implement similar amendments for S/RVs, such as Columbia Generating Station, Susquehanna Steam Electric Station, and River Bend Station, as delineated below. The change proposed herein to the JAF TS is consistent with these approved amendments.

1. Letter from L. J. Klos (U.S. Regulatory Commission) to M. E. Reddemann (Energy Northwest), "Columbia Generating Station - Issuance of Amendment Re: To Modify Technical Specifications Surveillance Requirements 3.4.3.1 and 3.4.4.1 Safety/Relief Valve Setpoint Lower Tolerance (CAC No. MF7699)," dated March 9, 2017 (ADAMS Accession No. ML17052A125)

2. Letter from B. K. Vaidya (U.S. Regulatory Commission) to T. S. Rausch (PPL Susquehanna, LLC), "Susquehanna Steam Electric Station, Units 1 and 2 - Issuance of Amendments Re: Change to Technical Specifications (TSs) Surveillance Requirements (SRs) 3.4.3.1 to Revise the Lower Surveillance Tolerances (TAC Nos. ME5050 and ME5051)," dated November 17, 2011 (ADAMS Accession No. ML11292A137)

3. Letter from M. Webb (U.S. Nuclear Regulatory Commission) to P. D. Hinnenkamp (Entergy Operations, Inc.), "River Bend Station, Unit 1 - Issuance of Amendment Re: Modification of the Technical Specification Surveillance Requirements for the Safety/Relief Valves (TAC No. MB5090)," dated February 13, 2003 (ADAMS Accession No. ML030450307)

4.3 No Significant Hazards Consideration

Amendment is proposed to the JAF TS SR 3.4.3.1, "Safety/Relief Valves (S/RVs)", to change the lower surveillance lift setpoint tolerance for the S/RVs. The current tolerance is based on ±3%. The proposed change will revise the lower setpoint surveillance tolerances from -3% to -5%. This change only applies to the as-found tolerance and not to the as-left tolerance, which will remain unchanged at ±1 %, as specified in TS SR 3.4.3.1, of the safety lift setpoint. The as-found tolerances are used for determining operability and to increase sample sizes for S/RV testing. No changes to the actual safety function lift setpoints are required for the valves installed in the plant. There will be no physical modifications to the valves.

CEG has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set
forth in 10 CFR 50.92, "Issuance of amendment", as presented below:

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

Response: No.

This change has no influence on the probability or consequences of any accident previously evaluated. The lower setpoint tolerance change does not affect the operation of the valves and it does not change the as-left setpoint tolerance. The change only affects the lower tolerance for valve opening and does not change the upper tolerance, which is the limit that protects from overpressurization.

The proposed amendment do not involve physical changes to the valves, nor do they change 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.

The proposed amendment do not change any other behavior or operation of any S/RVs, and, therefore, has no significant impact on reactor operation. They also have no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the UFSAR.

Based on the above, it is concluded that the proposed change to the S/RV surveillance requirement 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.

The proposed change to the S/RV safety lower setpoint tolerance from -3% to -5% only affects the criteria to determine when an as-found S/RV test is considered to be acceptable. This change does not affect the criteria for the upper setpoint tolerance.

The proposed lower setpoint tolerance change does not adversely affect the operation of any safety-related components or equipment. The proposed amendment do not involve physical changes to the S/RVs, nor do they change the safety function of the S/RVs. The proposed amendment do not require any physical change or alteration of any existing plant equipment. No new or different equipment is being installed, and installed equipment is not being operated in a new or different manner. There is no alteration to the parameters within which the plant is normally operated. This change does not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed, and no changes are being made to the procedures relied upon to respond to an off-normal event as described in the UFSAR. As such, no new failure modes are being introduced.

The change does not alter assumptions made in the safety analysis and licensing basis.

Therefore, the proposed changes do 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 proposed lower setpoint tolerance change only affects the criteria to determine when an as-found S/RV test is considered to be acceptable. This change does not affect the criteria for the S/RV setpoint upper setpoint tolerance. The TS setpoints for the S/RVs are not changed. The as-left setpoint tolerances are not changed by this proposed change and remain at ± 1% of the safety lift setpoint.

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 significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation.

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

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

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operations 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 do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meet 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

6.1 GE Nuclear Energy Licensing Topical Report NEDC-31753P, “BWROG In-Service Pressure Relief Technical Specification Revision Licensing Technical Report,” dated February 1990

6.2 NRC SE for GE Nuclear Energy Licensing Topical Report NEDC-31753P, March 8, 1993 (ADAMS Legacy Library Accession No. 9702070012, non-publicly available)

6.3 NRC SE for Issuance of Amendment for James A. FitzPatrick Nuclear Power Plant (TAC No. M89368), Technical Specification Amendment 217, dated September 28, 1994, ML01950384

6.4 James A. FitzPatrick Final Safety Analysis Report, Revision 28, April 2023

6.5 ASME, OM Code, 2004 Edition through 2006 Addenda

ATTACHMENT 2

License Amendment Request

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PROPOSED TECHNICAL SPECIFICATION MARKED-UP PAGES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoint of the required S/RVs is 1145 ~~±~~ + 34.3 ~~psig~~or – 57.2 psig. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.4.3.2	Verify each required S/RV is capable of being opened.	In accordance with the INSERVICE TESTING PROGRAM

ATTACHMENT 3

License Amendment Request

James A. FitzPatrick Nuclear Power Plant

Docket No. 50-333

PROPOSED TECHNICAL SPECIFICATION BASES MARK-UP PAGES

BASES (continued)

APPLICABLE SAFETY ANALYSIS

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Refs. 3 and 4). For the purpose of the analyses (Ref. 4), nine S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that nine S/RVs are capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig (at the vessel bottom) is met during the most severe pressurization transient.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 5 discusses additional events that are expected to actuate the S/RVs.

S/RVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 6).

LCO

The safety function of nine S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 3 and 4). 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).

The single nominal S/RV setpoint is established (Ref. 2) to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The single nominal S/RV setpoint is set below the RPV design pressure (1250 psig) in accordance with ASME Code requirements. The transient evaluations in Reference 5 are based on this single setpoint, but also include the additional uncertainties of + 3% or – 5%~~± 3%~~ of the nominal setpoint to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the analysis limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, nine S/RVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The S/RVs may be required to provide pressure relief to discharge energy 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 less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of the inoperable required S/RVs cannot be restored to OPERABLE status, 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 required S/RVs open at the pressures assumed in the safety analysis of References 3 and 4. The demonstration of the S/RV safety function 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% or – 5%~~± 3%~~ for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

(continued)

ATTACHMENT 4

License Amendment Request

James A. FitzPatrick Nuclear Power Plant

Docket No. 50-333

PROPOSED TECHNICAL SPECIFICATION CLEAN PAGES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoint of the required S/RVs is 1145 + 34.3 or – 57.25 psig. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.4.3.2	Verify each required S/RV is capable of being opened.	In accordance with the INSERVICE TESTING PROGRAM