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United States Nuclear Regulatory Commission
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Perry Nuclear Power Plant
Docket No. 50-440
License Amendment Request Pursuant to 10CFR50.90: Modification of the Safety Setpoint Requirements for the Safety Relief Valves

Ladies and Gentlemen:

Nuclear Regulatory Commission review and approval of a license amendment for the Perry Nuclear Power Plant (PNPP) is requested pursuant to 10CFR50.90. The proposed license amendment request increases the present ±1% tolerance on the safety mode lift setpoint for the safety/relief valves (SRVs) to ±3%. This change has been approved by the NRC staff on a generic basis as documented in NEDC-31753-P-A SER

Attachment 1 provides the Summary, a Description of the Proposed Technical Specification Change, a Safety Analysis, and an Environmental Consideration. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the annotated Technical Specification page reflecting the proposed change. The annotated Bases pages in Attachment 4 are provided for information only since the Bases are not a formal part of the Technical Specifications. Attachment 5 provides details of the plant specific analysis NEDC-32307P, "Safety Review for Perry Nuclear Power Plant Safety/Relief Valve Setpoint Tolerance Relaxation / Out-of-Service Analyses." This report is considered by General Electric (GE) to be proprietary information and an affidavit from GE to that effect is provided. Pursuant to 10CFR2.790 it is requested that the information contained in Attachment 5 be withheld from public disclosure.

If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Manager - Regulatory Affairs, at (440) 280-5606.

Very truly yours,

Lew W Myers

Attachments

(Attachment 5 contains 10CFR2.790 information. Upon removal of Attachment 5, the remainder of this package may be disclosed.)

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III
State of Ohio

I, Lew W. Myers, being duly sworn state that (1) I am Vice President - Nuclear, of the Centerior Service Company, (2) I am duly authorized to execute and file this certification on behalf of The Cleveland Electric Illuminating Company and Toledo Edison Company, and as the duly authorized agent for Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

Lew W. Myers

Lew W. Myers

Sworn to and subscribed before me, the 13th day of July, 1998

Jane E Mott

JANE E. MOTT
Notary Public, State of Ohio
My Commission Expires Feb. 20, 2000
(Recorded in Lake County)

SUMMARY

The Boiling Water Reactor Owners Group, with the assistance of General Electric, submitted Licensing Topical Report (LTR) NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," for Nuclear Regulatory Commission (NRC) review and approval (Reference 1). The LTR provided justification for the relaxation of safety/relief valve (SRV) safety mode (spring mode) lift setpoint tolerance from ±1% to ±3%. The NRC determined in its corresponding Safety Evaluation Report (NEDC-31753-P-A SER) that it was acceptable for licensees to submit Technical Specification amendment requests to revise lift setting tolerances to ±3% provided that the setpoints, for those SRVs tested, were restored to ±1% (Reference 2). The NRC SER instructed licensees implementing the Technical Specification modifications to provide plant specific analyses confirming the acceptability for revising lift setting tolerances to ±3%. These plant specific analyses and evaluations are summarized below and documented in NEDC-32307P "Safety Review for Perry Nuclear Power Plant Safety/Relief Valve Setpoint Tolerance Relaxation/Out-of-Service Analyses," dated May 1994 (Reference 3 and Attachment 5).

It is estimated that this change will save the Perry Nuclear Power Plant (PNPP) approximately $8.6 million over the remaining life of the plant. In addition, the proposed change will reduce personnel radiation exposure resulting from valve refurbishment and will eliminate additional testing which has no impact on plant safety. We request NRC Staff review and approval of this proposed license amendment by December 31, 1998 in order to support procedure changes necessary to implement this Amendment for the next refueling outage (RFO7) scheduled to start April 10, 1999.

The proposed license amendment request is similar to amendments previously approved by the NRC Staff for the James A. Fitzpatrick Nuclear Power Plant in an SER dated September 28, 1994; for LaSalle County Station, Unit 1 in an SER dated January 3, 1996, and for Grand Gulf Nuclear Station in an SER dated June 12, 1996.

Appropriate Updated Safety Analysis Report (USAR) changes will be completed to describe the change made to increase the SRV safety mode setpoint tolerance to ±3%. It should be noted that no changes are being proposed to the current opening setpoints for the SRVs in their relief mode (pneumatic mode) of operation.

DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION CHANGE

This proposed change increases the present ±1% tolerance on the safety mode lift setpoint for the SRVs to ±3%. Specifically, Surveillance Requirement (SR) 3.4.4.1 is being modified to revise the lift setpoints to reflect:

Setpoint (psig)
1165 ± 34.9
1180 ± 35.4
1190 ± 35.7

The annotated page for the proposed change to SR 3.4.4.1 is provided in Attachment 3.

Additionally, associated Bases changes are included in Attachment 4 for information only, since Bases are not a formal part of the Technical Specifications (Bases changes are processed per the Technical Specification Bases Control Program, Specification 5.5.11).

SAFETY ANALYSIS

The American Society of Mechanical Engineers (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 nuclear pressure relief system, the size and number of SRVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The SRV setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure (1375 psig - also referred to as the "upset limit"). The overpressure transient evaluation in USAR Chapter 15 is based on these setpoints and includes additional uncertainties of ±1% of the nominal setpoint to account for potential setpoint drift. This provides an additional degree of conservatism.

The LTR NEDC-31753P provided justification for the relaxation of SRV safety mode lift setpoint tolerance from ±1% to ±3% as denoted in SR 3.4.4.1. The NRC determined in its corresponding Safety Evaluation Report that each licensee choosing to implement the SRV setpoint tolerances should provide a plant specific analysis that includes the following:

1) Transient analysis, using NRC approved methods, of the abnormal operational occurrences as described in NEDC-31753P, should be performed utilizing a ±3% setpoint tolerance for the safety mode of the SRVs.

2) Analysis of the design basis overpressurization event using the 3% tolerance limit for the SRV setpoint is required to confirm that the vessel pressure does not exceed the ASME pressure vessel code upset limit.

3) The plant specific analyses described in Items 1 and 2 should assure that the number of SRVs included in the analyses corresponds to the number of valves required to be operable in the Technical Specifications.

4) Re-evaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the 3% tolerance limit.

5) Evaluation of the ±3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

6) Evaluation of the effect of the 3% tolerance limit on the containment response during loss of coolant accidents and the hydrodynamic loads on the SRV discharge lines and containment should be completed.

The plant specific analyses and evaluation requested above have been completed and are provided in Attachment 5. Since this is considered a proprietary document, a summary of results is provided below. In addition to the analyses in Attachment 5 (portions of which used PNPP Cycle 5 information), the following summary also notes the applicable results of the current Cycle 7 analyses.

1. TRANSIENT ANALYSIS

Limiting transient events for PNPP Cycle 5 were evaluated (Reference 3) to determine whether the increase in SRV safety (spring) mode setpoint tolerance from ±1% to ±3% would be acceptable. The Cycle 5 evaluation identified the Misoriented Bundle event as the most limiting event for the operating limit minimum critical power ratio (MCPR). The proposed change to the SRV tolerances does not affect the MCPR calculation for the Misoriented Bundle event since there is no SRV actuation in the sequence of events. The evaluation of ΔCPR also included several pressurization transient events (i.e., events in which the SRVs actuate) such as the Generator Load Rejection with no Bypass and the Feedwater Controller Failure. These pressurization events were not affected since credit is taken for SRV operation in the relief mode (pneumatically opened mode of SRV operation), thus, MCPR is not dependent on the SRV safety mode of operation. Also, the safety mode setpoint relaxation is not a factor since the SRVs open after the occurrence of

MCPR during transients. Therefore, the change in the safety mode setpoint does not affect these transient results.

A review of the current Cycle 7 results (USAR 15B.15) identified the Load Reject without Bypass event as the most limiting event for the operating limit MCPR. The proposed change to the SRV tolerances does not affect the MCPR calculation for the Load Reject without Bypass event since there is no impact on MCPR once the reactor scram has occurred. The limiting MCPR occurs prior to the opening of any SRV, and thus the limiting MCPR is not affected.

2. OVERPRESSURE PROTECTION ANALYSIS

The ASME Code requires the peak vessel pressure to remain less than the upset limit of 1375 psig during the limiting overpressure event. For PNPP, this is a Main Steam Isolation Valve (MSIV) closure followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV closure).

An analysis was performed for this event (Reference 3 - Cycle 5 inputs) assuming only 13 of the 19 SRVs installed at PNPP function. The analysis assumed a setpoint tolerance of ±3%, and a 102% power and 105% flow condition with conservative end-of-cycle nuclear dynamic parameters. This analysis ensures the operability requirements of LCO 3.4.4, "Safety/Relief Valves (S/RVs)" are appropriate. For Cycle 5, the peak main steamline pressure was 1264 psig and the peak vessel bottom pressure was 1294 psig.

A review of Cycle 7 results (USAR 15B.5.2.2), which also included the ±3% safety mode setpoint tolerance, identified the peak main steamline pressure as 1258 psig and the peak vessel bottom pressure as 1289 psig. These pressures provide significant margins to the ASME Code upset limit of 1375 psig. Therefore, PNPP satisfies the ASME limit with a ±3% SRV setpoint tolerance using only 13 SRVs. Since the overpressure protection analysis is cycle-specific, the peak vessel pressure is demonstrated every operating cycle to remain below the ASME criteria for overpressure protection.

3. NUMBER OF SRVs USED IN ANALYSIS

The overpressure protection analysis, described in Item 2 above, assumes only the Technical Specification required 13 SRVs are functioning. This is consistent with the Bases for LCO 3.4.4, "Safety/Relief Valves (S/RVs)." The Technical Specification requirements are based on this overpressure protection analysis, not the transient analysis described in Item 1. The plant specific transient analyses, described in Item 1 above for the abnormal operational occurrences, assume that 17 out of 19 SRVs are functioning (USAR 15B.0.1).

4. HIGH PRESSURE SYSTEMS

The impact of the increased setpoint tolerance on the safety functions of the High Pressure Core Spray (HPCS) System, the Reactor Core Isolation Cooling (RCIC) System, and the Standby Liquid Control System (SLCS) was evaluated. The most significant impact is the increased reactor pressure specified for operation. Note that the proposed change does not affect the relief mode setpoints, which remain at:

Setpoint (psig)	
1103 ± 15	(1 valve)
1113 ± 15	(9 valves)
1123 ± 15	(9 valves)

High Pressure Core Spray System

The HPCS System is designed to deliver water to the reactor vessel at ≥ 517 gpm, with the reactor vessel pressure 1177 psig (1165 + 11.6 psig) above the pressure at the source of suction. The increase in SRV setpoint tolerance increases the maximum reactor pressure for HPCS System injection to 1200 psig (1165 + 34.9 psig). A review of the pump performance curves indicates that the HPCS System has sufficient margin to deliver flow in excess of 517 gpm at 1200 psig. The HPCS pump design has a discharge rating of 1575 psig. This pressure is well above the pressures which may result from SRV safety mode setpoint tolerance relaxation. Therefore, the increase in SRV setpoint tolerance has been determined to be acceptable for the HPCS System.

Reactor Core Isolation Cooling System

The design requirements for the RCIC System, including the RCIC turbine, is to deliver 700 gpm in the reactor pressure range of 150 to 1177 psig. The increase in SRV setpoint mode tolerance requires the RCIC System to deliver 700 gpm in the reactor pressure range of 150 to 1200 psig. A review of the pump performance curves indicates that the RCIC System has sufficient margin to deliver flow in excess of 725 gpm at 1200 psig. The RCIC pump design has a discharge rating of 1575 psig. This pressure is well above the pressures which may result from SRV safety mode setpoint tolerance relaxation. Therefore, the increase in SRV setpoint tolerance has been determined to be acceptable for the RCIC System.

A review of the RCIC turbine performance curves indicates that the turbine has the capacity to develop the horsepower and speed required by the pump for the increased pressure conditions. The turbine speed must be raised from 4550 rpm to 4600 rpm. This is below the maximum permitted turbine shaft speed of approximately 5000 rpm. The increased turbine rated speed reduces the overspeed trip margin from 125% to 122.3%. However, this reduction in margin was found to be acceptable in accordance with the GE Service Information Letter No. 377. The steam flow increase to drive the turbine to a higher speed is less than 2%. This steam flow increase and a small increase in turbine exhaust pressure are not expected to have any adverse impact to the design life and performance of the turbine system. The design pressure of the RCIC System steam supply lines and turbine was based on the reactor design pressure of 1250 psig. This pressure is above the pressures which may result from SRV safety mode setpoint tolerance relaxation. Therefore, the increase in SRV setpoint tolerance has been determined to be acceptable for the RCIC System.

Standby Liquid Control System

The SLCS was evaluated in Reference 3 based on the SRV safety mode settings. However, SLCS operation is not affected by the SRV setpoint tolerance increase. The pressure used for system performance is based on the SRV relief settings of the system, not the SRV safety settings. This proposed change does not affect the relief settings of the SRVs.

Other Components (MOVs, Associated Piping)

The minimum design pressure of the piping, valves and components which are part of the primary reactor coolant pressure boundary is 1250 psig. Since the design pressure is higher than the lowest opening pressure of the SRV safety mode at 1200 psig, an increase in SRV setpoint tolerance has been determined to be acceptable.

5. ALTERNATE OPERATING MODES

The alternate operating modes, including Maximum Extended Operating Domain (MEOD), Increased Core Flow Region, and Single Loop Operation (SLO) were considered in determining the most restrictive analytical conditions (i.e., the most limiting operating mode) for performing the analyses associated with this proposed Technical Specification change (see Items 1 and 2, above).

6. LOSS OF COOLANT ACCIDENT (LOCA) PERFORMANCE

LOCA

The LOCA analysis was reviewed to determine the effect of an increase in SRV setpoint tolerance on ECCS performance. The ECCS is designed to maintain fuel integrity, during postulated LOCAs, below 10CFR50.46 limits. A change in SRV opening pressure can only affect the containment pressure response for LOCAs in which SRV actuations occur. No SRV actuation occurs for large pipe breaks inside the containment because the vessel depressurizes through the break. Therefore, an increase in SRV opening pressure will not affect the results of the design basis accident (DBA) LOCA evaluated in the USAR. For a double-ended guillotine break of one main steamline outside the containment, the reactor vessel is isolated upon closure of the MSIVs. The peak pressure of this accident is bounded by the overpressure protection event. In addition, 10CFR50.46 calculations of peak clad temperatures for this event and for small-break LOCAs are insensitive to SRV actuation. For small break LOCAs, the SRVs are armed with low-low set logic (LLS) allowing the relief mode of 6 SRVs to relieve reactor pressure to below 1000 psig. Once the logic is initiated, the opening and closing setpoints of these SRVs are automatically reset to lower values by the LLS logic. This logic is not affected by the setpoint tolerance change since it acts on the relief mode of SRV actuation and not on the safety mode of operation. Consequently, since PNPP is a LLS plant, the peak pressure effect from the SRV setpoint relaxation on a small break LOCA event is negligible.

Containment Response

The most limiting LOCA event in terms of peak containment pressure, temperature and peak suppression pool temperature is a DBA LOCA. Relaxation of the SRV setpoint tolerance has no effect on this limiting event because the vessel depressurizes without any SRV actuation.

LOCA-Related Hydrodynamic Loads

The LOCA hydrodynamic loads, such as pool swell, condensation oscillation and chugging, are all dependent on peak containment responses during a DBA LOCA and therefore are not affected by the increase in the SRV safety mode setpoint tolerances.

SRV Discharged Loads

Steam discharged from the SRVs is routed through the SRV discharge lines and through the quencher into the suppression pool. Actuation of SRVs introduces high pressure steam which quickly pressurizes the discharge piping resulting in forced expulsion of the water as well as air initially in the piping. The SRV loads resulting from SRV actuation includes thrust loads, air-clearing loads, reaction loads, and air bubble loads impacting SRV piping, piping anchors, quenchers, and submerged structures (USAR Appendix 3B). An increase of the SRV setpoint tolerance will result in a corresponding increase in discharge flow rate into the discharge piping. At 1%, the maximum pressure setting can be as high as 1202 psig. At 3%, the maximum pressure setting can be as high as 1226 psig. The maximum increase is only 24 psig or about 2 %. The effect of SRV discharged loads from the SRV safety mode setting tolerance increase can be evaluated as follows:

- SRV Discharge Piping to the First Anchor - The original loads for this portion of piping were generically derived for all BWR/6 plants and were based on worst case input conditions. The resultant SRV load was developed using an SRV flow rate input higher than the maximum expected SRV flow rate (with 3% setpoint tolerance) for PNPP. As indicated in USAR Table 3.9.3 (k), the resultant stresses for this portion of SRV piping show large safety margins in excess of 40%.

- Quencher Loads - PNPP uses the standard GE X-quencher design. The generic loads defined for the original quencher loading design were very conservative and therefore were determined to be unaffected by the increase in SRV setpoint tolerance.

- Submerged Structure and Pool Boundary Loads - The loads on the submerged structures are based on the peak air bubble pressures determined with the generic X-quencher methodology. The correlation which was used to determine the increase in the SRV generic peak bubble pressure due to the increase in the SRV discharge flow rate was determined to be very conservative. Therefore there is no effect on the submerged structures load definition.

ATWS Mitigation Capability

An Anticipated Transient Without Scram (ATWS) is a beyond design basis event. The potential impact of the SRV tolerance setpoint relaxation on ATWS performance is the compliance to vessel overpressure criteria of 1500 psig. The limiting event for this ATWS condition is the MSIV closure transient (assumes that a reactor scram does not occur on a reactor protection system signal). For this event, the initial reduction of power occurs from the ATWS high dome pressure recirculation pump trip, accompanied by the boron injection from the SLCS. This leads to an eventual reactor shutdown. During this transient, SRVs actuated accordingly. An analysis was performed with the reactor at 100% power and 100% recirculation flow. Assuming 2 SRVs failed to open on demand, the peak reactor vessel bottom pressure was calculated to be 1344 psig. This is significantly less than the acceptance limit, which is the ASME service level C (Emergency) value of 1500 psig. Therefore, the setpoint tolerance relaxation on the safety mode lift setpoint does not adversely impact the results of any ATWS event.

7. CONCLUSION

The analyses and evaluations support the relaxation of the as-found SRV safety mode lift setpoint tolerance from the current ±1% to ±3%. The analyses and evaluations summarized above have no significant safety impact on ECCS/LOCA performance, high pressure system performance, containment structural integrity, and ATWS analysis results. The analyses examined cycle dependent safety concerns, such as vessel overpressure margin and thermal limits, and demonstrated that the SRV safety mode tolerance combined with 2 SRVs out-of-service has no adverse impact upon plant safety for transient events. In addition, there is no adverse impact upon plant safety with 6 SRVs out-of-service for the vessel overpressure analysis. Future cycle-specific reload licensing evaluations verify continued applicability of the results from this analysis.

8. REFERENCES

1. NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," GE Nuclear Energy, dated February 1990.

2. NRC letter (NEDC-31753-P-A SER) from A.S. Thadani to C.L. Tully, Chairperson, BWR Owners' Group, "Acceptance for Referencing of Licensing Topical Report NEDC-31753P," dated March 8, 1993.

3. NEDC-32307P, "Safety Review for Perry Nuclear Power Plant Safety/Relief Valve Setpoint Tolerance Relaxation / Out-of-Service Analyses," GE Nuclear Energy, dated May 1994.

ENVIRONMENTAL CONSIDERATION

The proposed Technical Specification change request was evaluated against the criteria of 10CFR51.22 for environmental considerations. The proposed change does not significantly increase individual or cumulative occupational radiation exposures, does not significantly change the types or significantly increase the amount of effluents that may be released off-site and, as discussed in Attachment 2, does not involve a significant hazards consideration. Based on the foregoing, it has been concluded that the proposed Technical Specification change meets the criteria given in 10CFR51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement.

COMMITMENTS WITHIN THIS LETTER

The following table identifies those actions which are considered to be regulatory commitments. Any other actions discussed in this document represent current or planned actions and are described for the NRC's information. Please notify the Manager - Regulatory Affairs at the Perry Nuclear Power Plant of any questions regarding this document or any associated regulatory commitments.

Commitments
Appropriate Updated Safety Analysis Report (USAR) changes will be completed to describe the change made to increase the SRV safety mode setpoint tolerance to ±3%.

SIGNIFICANT HAZARDS CONSIDERATION

The standards used to arrive at a determination that a request for amendment involves no significant hazards considerations are included in the Commission's Regulation, 10CFR50.92, which states that the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The proposed amendment has been reviewed with respect to these three factors and it has been determined that the proposed change does not involve a significant hazard because:

(1) The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change allows an increase in the as-found safety relief valve (SRV) safety mode setpoint tolerance, determined by test after the valves have been removed from service, from ±1% to ±3%. The proposed change does not alter the Technical Specification requirements on the nominal SRV safety mode lift setpoints, the SRV relief mode setpoints, the required frequency for the SRV lift setpoint tests, or the number of SRVs required to be operable. This change does not involve physical changes to the SRVs, nor does it change the operating characteristics or safety function of the SRVs.

Consistent with current requirements, this change continues to require that the SRVs be adjusted to within ±1% of their nominal lift setpoints following testing. This change does not change the behavior and operation of any SRV and therefore has no significant impact to reactor operation. It also has no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the Updated Safety Analysis Report. In addition, this change does not change SRV actuation. Therefore, this change will not increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," and were reviewed and approved by the NRC. The plant specific evaluations, required by the NRC's Safety Evaluation for NEDC-31753P and performed to support this proposed change, are contained in NEDC-32307P, "Safety Review for PNPP Safety/Relief Valve Setpoint Tolerance Relaxation / Out-of-Service Analyses," dated May 1994. These analyses and evaluations show that there is adequate margin to the design core thermal limits and to the reactor vessel pressure limits using a ±3% SRV setpoint

tolerance. They also show that operation of the high pressure injection systems will not be adversely affected; and the containment response from a loss of coolant accident will be acceptable.

Therefore, this change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

(2) The proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change to allow an increase in the SRV safety mode setpoint tolerance from ±1% to ±3% does not alter the nominal SRV lift setpoints or the number of SRVs required to be operable. This change does not involve physical changes to the SRVs, nor does it change the operating characteristics or the safety function of the SRVs. The proposed change does not involve a physical alteration of the plant. No new or different equipment is being installed. The proposed change does not impact core reactivity nor the manipulation of fuel bundles. There is no alteration to the parameters within which the plant is normally operated. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) The proposed change will not involve a significant reduction in the margin of safety.

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. The proposed change does not significantly impact any safety analysis assumptions or results.

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

Based on the above considerations, it is concluded that a significant hazard would not be introduced as a result of this proposed change. Also, since NRC approval of this change must be obtained prior to implementation, no unreviewed safety question can exist.