PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

PSEG
Nuclear LLC

APR 03 2015

10 CFR 50.90

LR-N15-0021
LAR S15-02

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

Salem Nuclear Generating Station Units 1 and 2
Renewed Facility Operating License Nos. DPR-70 and 75
NRC Docket Nos. 50-272 and 50-311

Subject: **License Amendment Request Regarding Replacement of Source Range and Intermediate Range Neutron Monitoring Systems**

In accordance with 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment to Renewed Facility Operating License Nos. DPR-70 and 75 for Salem Nuclear Generating Station, Units 1 and 2. In accordance with 10 CFR 50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed changes revise Technical Specification (TS) 3/4.3.1,"Reactor Trip System Instrumentation," to support planned plant modifications to replace the existing source range and intermediate range nuclear instrumentation with equivalent neutron monitoring systems to increase system reliability.

There are no regulatory commitments contained in this letter.

Attachment 1 provides an evaluation supporting the proposed changes.
Attachment 2 contains marked-up TS pages to indicate the proposed changes.
Attachment 3 provides the proposed changes to the TS Bases for information only.

PSEG requests NRC approval of the proposed License Amendment by April 3, 2016, to support planned plant modifications to replace the existing source range and intermediate range nuclear instrumentation. PSEG requests a staggered implementation with Salem Unit 1 implementation during the spring 2016 refueling outage (1R24) and the Salem Unit 2 implementation during the spring 2017 refueling outage (2R22).

These proposed changes have been reviewed by the Plant Operations Review Committee.

If you have any questions or require additional information, please contact Brian Thomas at (856) 339-2022.

APR 03 2015

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

Executed on APR 03 2015
(Date)

Respectfully,

John F. Perry

John F. Perry
Site Vice President
Salem Generating Station

Attachments:

1. License Amendment Request Regarding Replacement of Source Range and Intermediate Range Neutron Monitoring Systems
2. Technical Specification Proposed Changes (mark-up pages)
3. Technical Specification Bases Proposed Changes (for information only)

cc: Mr. D. Dorman, Administrator, Region I, NRC
Ms. C. Sanders-Parker, Project Manager, NRC
NRC Senior Resident Inspector, Salem
Mr. P. Mulligan, Manager IV, NJBNE
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. T. Cachaza, Salem Commitment Tracking Coordinator

SALEM GENERATING STATION
RENEWED FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75
DOCKET NO. 50-272 AND 50-311

License Amendment Request Regarding Replacement of Source Range and Intermediate Range Neutron Monitoring Systems

Table of Contents

1. DESCRIPTION 2
2. PROPOSED CHANGE 2
3. BACKGROUND 3
4. TECHNICAL ANALYSIS 4
5. REGULATORY ANALYSIS 10
   5.1 No Significant Hazards Consideration 10
   5.2 Applicable Regulatory Requirements/Criteria 12
   5.3 Precedent 13
6. ENVIRONMENTAL CONSIDERATION 13
7. REFERENCES 13

1.0 DESCRIPTION

The proposed changes revise Technical Specification (TS) 3/4.3.1,"Reactor Trip System Instrumentation," to support planned plant modifications to replace the existing source range (SR) and intermediate range (IR) nuclear instrumentation with equivalent neutron monitoring systems. Due to reliability and parts obsolescence issues, the Westinghouse SR and IR neutron flux monitors (original plant installed equipment) are being replaced with the Thermo Scientific 300i Neutron Flux Monitoring Systems. The Thermo Scientific equipment is equivalent with respect to interface with the rest of the Nuclear Instrumentation System (NIS) and Reactor Trip System (RTS), and meets all the functional requirements of the Westinghouse equipment being replaced.

2.0 PROPOSED CHANGE

The proposed changes are described below and indicated on the marked-up TS pages provided in Attachment 2 of this submittal. Proposed changes to the TS Bases are provided in Attachment 3 for information only. Changes to the affected TS Bases pages will be incorporated per TS 6.17 (Unit 1) and TS 6.16 (Unit 2), "Technical Specifications (TS) Bases Control Program."

1. TS 3/4.3.1, Table 2.2-1

   Change the Functional Unit 5 (Intermediate Range, Neutron Flux) Allowable Value from ≤ 30% of RATED THERMAL POWER to ≤ 38.5% of RATED THERMAL POWER.

   Change the Functional Unit 6 (Source Range, Neutron Flux) Allowable Value from ≤ 1.3×10^5 counts per second to ≤ 1.44×10^5 counts per second.

2. TS 3/4.3.1, Table 3.3-1

   Delete - # # for Functional Unit 6.A (Source Range, Neutron Flux – Startup)

3. TS 3/4.3.1, Table 3.3-1 (Continued), Page 3/4 3-5

   Delete - # # *High voltage to detector may be de-energized above P-6.*

4. TS 3/4.3.1, Table 3.3-1 (Continued), Page 3/4 3-7

   Change the permissive P-6 reset from < 6×10^{-11} amps to < 4.7×10^{-6} % of RATED THERMAL POWER.

5. TS 3/4.3.1, Table 4.3-1

   Add two new footnotes (# and # #) that apply to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for Functional Units 5 and 6. The Allowable Value changes indicated above affect these Functional Units. The footnotes read as follows:

   \# If the as-found channel setpoint is outside its predetermined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

\# # The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the nominal Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the nominal Trip Setpoint are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the Technical Specification Bases.

6. Revise the Bases Index page to reflect the page number change.

3.0 BACKGROUND

The RTS consists of all components from the field-mounted process instrumentation (e.g., transmitters, RTDs, neutron detectors) to the reactor trip switchgear, whose functioning initiates a reactor trip when required. Salem Updated Final Safety Analysis Report (UFSAR) Section 7.2 states that the RTS includes the NIS, process control system, and the Solid State Protection System (SSPS).

The primary function of the NIS is to protect the reactor core by monitoring neutron flux and generating appropriate trips and alarms for various phases of reactor operating and shutdown conditions. It also indicates reactor status during startup and power operation. The NIS consists of three discrete but overlapping ranges. They are the SR, IR and power range (PR). Reactor startup and power escalation requires a permissive signal from the higher range instrumentation channels before the operator can manually block the lower range reactor trips.

The PR neutron flux trip circuit initiates a reactor trip when two of the four PR channels exceed the Trip Setpoint. There are two bistables per PR channel used for a high and low reactor trip setting. The PR neutron flux - high setting reactor trip provides protection during normal power operation and is always active (i.e., cannot be blocked). The PR neutron flux - low setting reactor trip, which provides protection during startup, can be manually blocked when two out of the four PR channels indicate above approximately 10 percent power (permissive P-10). Three out of the four PR channels below P-10 automatically reinstates the PR neutron flux - low setting reactor trip.

The IR neutron flux trip circuit initiates a reactor trip when one out of two IR channels exceeds the Trip Setpoint. This reactor trip, which provides protection during reactor startup, can be manually blocked when two out of the four PR channels are above P-10. Three out of the four PR channels below this value automatically reinstates the IR neutron flux reactor trip.

The SR neutron flux trip circuit initiates a reactor trip when one out of two SR channels exceeds the Trip Setpoint. This reactor trip, which provides protection during reactor startup, can be manually blocked when one out of two IR channels reads above the permissive P-6 setpoint value, and is automatically reinstated when both IR channels decrease below P-6. This reactor trip is automatically blocked by the P-10 permissive. The SR neutron flux reactor Trip Setpoint is established between the P-6 setpoint and the upper range of the SR scale.

The PR neutron flux- low setting reactor trip, IR neutron flux reactor trip, and SR neutron flux reactor trip described above are designed to protect the reactor core against power excursions during reactor startup or low power operation. The SR and IR neutron flux reactor trips provide redundant protection to the PR neutron flux – low setting reactor trip. No credit is taken for the reactor trips associated with either the SR or IR channels in the accident analyses described in

Chapter 15 of the Salem UFSAR. Their functional capability enhances the overall reliability of the Reactor Protection System (RPS).

Due to reliability and parts obsolescence issues, the existing Westinghouse SR and IR neutron monitoring systems are being replaced with the Thermo Scientific 300i Neutron Flux Monitoring System. The existing SR and IR detectors use boron triflouride (BF3) and compensated ion chambers respectively. Thermo Scientific detectors use a fission chamber that performs both the SR and IR monitoring functions. The Thermo Scientific detectors have a 40-year design life, eliminating the need to periodically replace the limited life SR BF3 detector assemblies and IR compensated ion chamber detector assemblies.

4.0 TECHNICAL ANALYSIS

1. Overview

   A. The Thermo Scientific equipment is equivalent to the existing Westinghouse SR and IR instrumentation with respect to interface with the rest of the NIS and RTS, and meets all the functional requirements of the Westinghouse equipment being replaced. Both systems are classified as safety-related (Class 1E). The Thermo Scientific instrumentation does differ from the Westinghouse SR and IR instrumentation in the following aspects:

      - Detector Orientation – The existing SR BF3 detector is positioned below the centerline of the core height. The Thermo Scientific detector assembly utilizes two fission chambers to provide both the SR and IR signals. The fission chamber detector will be positioned such that the centerline of the sensitive volume aligns with the centerline of core height.
      - Source Range Scale – The SR indication scale changes from 10^0–10^6 counts per second (cps) (six decades) to 10^{-1}–10^6 cps (seven decades).
      - Source Range High Flux at Shutdown Alarm – The existing SR instrumentation has a setpoint of 0.5 to 1.0 decade above background SR level. The Thermo Scientific alarm setpoint is electronically established based on a selectable fixed ratio between 1.25 to 4.0 times steady-state, and is automatically reduced as steady-state count rate decreases.
      - Source Range De-energization – With the existing Westinghouse system, the SR indication is disabled by de-energizing high voltage to the detectors when the SR reactor trip is manually blocked upon receipt of the permissive P-6. This prevents damage to the BF3 detectors from operation beyond their design limits. Removing high voltage to the Thermo Scientific fission chamber detectors is not required; they remain energized through all levels of operation.
      - Intermediate Range Scale Units – The IR indication scale units change from amps to percent power.
      - Intermediate Range Scale – The IR indication scale changes from 10^{-11}–10^{-3} amps (eight decades) to 10^{-8}– 200% Rated Thermal Power (RTP) (ten decades).

   These differences do not affect the NIS reactor trip protective function.

B. Due to the changes in the IR detector output and units, an assessment was completed to verify adequate coordination between the P-6 setpoint and the SR neutron flux reactor trip setpoint for the Thermo Scientific instrumentation.

The SR neutron flux reactor trip setpoint and the P-6 setpoint are set relative to the overlap between the SR and IR scales. The P-6 setpoint is selected such that its bistable trips after the IR indication comes on scale (allows verification of IR operation) and before the SR indication goes off scale (within the overlap region of the instruments). The SR neutron flux reactor trip setpoint is established between the P-6 setpoint and the upper range of the SR scale, sufficiently above the P-6 value to allow the operator time to block the SR neutron flux reactor trip.

The Westinghouse IR P-6 setpoint ($1x10^{-10}$ amps) provides one decade overlap. The Thermo Scientific instrumentation extends the IR bottom end range two additional decades. The equivalent P-6 setpoint of 10^{-5} % RTP provides three decades of overlap. The assessment determined that there is margin between the SR neutron flux trip descending setpoint ((-) as-found tolerance) and the IR P-6 ascending setpoint ((+) as-found tolerance). This allows the operator sufficient time to actuate the SR neutron flux reactor trip block signal, and at the same time ensures a conservative signal overlap with the IR indication. The assessment also calculated a P-6 reset value of < $4.7x10^{-6}$ % RTP.

2. Calculation Assessment Results

Introduction

At the time Salem Nuclear Generating Station, Units 1 and 2 were designed, there were no industry standards, regulatory approved criteria, or guidelines that described how instrument uncertainties or setpoints were calculated. Regulatory Guide (RG) 1.105, Revision 1, "Instrument Setpoints," was published in November 1976 in response to the large number of reported instances in which instrument setpoints in safety-related systems drifted outside the limits specified in the TS. The single most prevalent reason for the drift of a measured parameter out of compliance with the TS was the selection of a setpoint that did not allow sufficient margin between the setpoint and the TS limit. In March of 1977, the NRC requested that several utilities with Westinghouse Nuclear Steam Supply Systems (NSSS) reply to a series of questions concerning the methodology for determining instrument setpoints. Westinghouse developed a revised methodology in response to those questions, and published a document in June 1978 providing information regarding the instrument uncertainties assumed for justifying the RPS and Engineered Safety Features Actuation System (ESFAS) trip setpoints. Instrument Society of America (ISA) Standard S67.04 was subsequently prepared to provide the nuclear industry with guidelines on how to address instrument drift problems and their associated impact on plant setpoints. RG 1.105, Revision 2 (Reference 1) endorsed ISA S67.04-1982 (Reference 2) for use in establishing and maintaining setpoints in safety-related systems. The methodology used to calculate the Salem trip setpoints is consistent with ISA-S67.04-1982.

Westinghouse prepared a Salem plant specific setpoint methodology for protection systems in 1989 using the methodology in existence at that time (Reference 3). A Salem plant specific methodology bases document was issued in 1994 (Reference 4) providing a compilation of the calculations, terms, references, and assumptions made by

Westinghouse in the performance of the uncertainty calculations for the Salem plant setpoint study. Salem Technical Standard SC.DE-TS.ZZ-1904(Q) (Reference 5), which references the Westinghouse Salem setpoint methodology, provides the technical criteria for determining Salem instrument setpoints.

A setpoint calculation assessment was performed in support of the planned plant modifications to replace the existing SR and IR instrumentation. The results required changes to the associated values listed in TS 3/4.3.1 Tables 2.2-1 and 3.3-1, as described in Section 2.0 above.

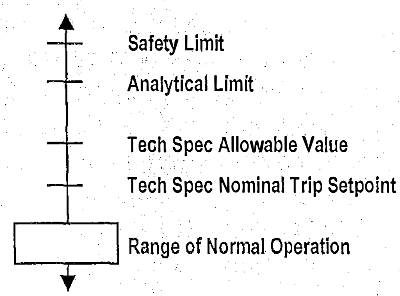
Basic Methodology

The loop uncertainty methodology is primarily based on the Square Root Sum of the Squares (SRSS) technique. The SRSS is a statistical method of combining multiple independent random errors in order to establish the total error attributable to all of the individual errors. The SRSS method accounts for the individual probabilities of random errors. This approach is valid where no dependency is present. The SRSS method of combining random error terms is a technique accepted by the NRC (Reference 1).

Random dependent and bias uncertainty terms must be addressed through a combination of the SRSS technique and algebraic summation. The individual random error terms are combined by SRSS to establish a single, resultant random error component. Algebraic summation is then used to combine all non-random (bias) terms to establish single positive and negative bias error components. The total error or uncertainty is obtained by combining the random and bias components.

The calculation results expressed in percent of span is percent of Equivalent Linear Full Scale (% ELFS, % span).

Evaluation of setpoint acceptability requires comparison of the total loop uncertainty against the operational ranges and the protected limits (process, analytical and/or safety limits). This setpoint relationship is based on guidance specified in RG 1.105. The typical reactor trip setpoint relationship is depicted as follows:



Safety Limits (SL) are the values chosen to reasonably protect the integrity of physical barriers that guard against the uncontrolled release of radioactivity. Analytical Limits (AL) typically are values utilized in the safety analyses, which were specifically chosen to allow plant equipment sufficient time to act and prevent exceeding the SLs. The Allowable Value (AV) represents an acceptable benchmark (specified by TS) that periodic calibrations/checks must fall within to ensure operability.

Total Loop Uncertainty (TLU)

TLU is determined by statistically combining the rack uncertainties and process measurement accuracy (PMA). The methodology of combining the uncertainty random terms with algebraic summation of bias terms has not changed. The proposed changes will implement values based on the design capabilities of the Thermo Scientific instrumentation.

The calculated TLUs are as follows:

SR ± 10.25 % span
IR + 2.01% span, - 3.69% span

Analytical Limit (AL)

The analyses establish specific limits and assumptions on plant design and operation. These limits, and their associated assumptions, form the bases of the instrument setpoint determinations and are known as AL. The values used within safety analyses, or determined from safety analyses, represent the AL that the individual protective actions must not exceed. The AL are documented within Salem UFSAR Table 15.1-3 "Trip Points and Time Delays to Trip Assumed in Accident Analysis," Westinghouse Setpoint Methodology for Protection Systems (Reference 3), or applicable PSEG calculations.

The SR and IR neutron flux reactor trips are not explicitly credited in the accident analyses so there is not a defined AL. However, the following values were used as ALs in the setpoint calculation assessment to determine if the existing reactor trip setpoints remain acceptable:

SR 1×10^6 cps
IR 75% RTP

Limiting Safety System Setting (LSSS)

The LSSS are the setpoints established for the plant's automatic safety systems. They take into account the required allowances between SLs and actual plant setpoints, to ensure that the SLs are not exceeded. A LSSS is chosen to begin protective action before the AL is reached to ensure that the consequences of a design basis accident are not more severe than the safety analysis predicted. The LSSS is comprised of two components – the Trip Setpoint and Allowable Value. The LSSSs (Trip Setpoints and Allowable Values) are specified in Salem TS Table 2.2-1.

Nominal Trip Setpoint

Nominal Trip Setpoints are the predetermined values at which the bistables are set, ensuring that the plant's automatic safety systems actuate prior to the process variable reaching the AL. The Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for the required allowances (e.g., calibration tolerances, instrumentation uncertainties, instrument drift) between the SL and actual plant setpoints to ensure that the SLs are not exceeded. The Trip Setpoint is considered to be adjusted consistent with the nominal value when the as-left setpoint is within the band allowed for channel calibration accuracy.

The following equation represents an acceptable method for determining the nominal Trip Setpoint.

Calculated Nominal Trip Setpoint = AL ± TLU

The TLU is summed or subtracted from the AL depending on whether the process is increasing or decreasing toward the nominal Trip Setpoint.

The calculated nominal Trip Setpoints were evaluated against the current SR and IR nominal Trip Setpoints. Positive margin existed for both the SR and IR trip setpoints, therefore the current TS nominal Trip Setpoints remain acceptable and unchanged.

Allowable Value (AV)

The AVs accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated. The AV defines a limit that the nominal Trip Setpoint must be maintained within to show that the uncertainties that are present within the loop, when it is periodically tested or calibrated, are consistent with the values used within its uncertainty/setpoint calculation. Operation with setpoints less conservative than the nominal Trip Setpoint but within the AV is acceptable because an allowance has been made in the safety analysis to accommodate this error.

The Thermo Scientific instrumentation accuracy is better than the existing Westinghouse instrumentation; however, the Thermo Scientific detectors have a wider detection range as compared to the Westinghouse detectors. This results in an increase in the SR and IR AV.

The calculated AVs are as follows:

SR	$\leq 1.44\ 10^5$ cps
IR	$\leq$ 38.5% RTP

As-Found Tolerance

Because all devices experience drift (an undesirable change in output over a period of time unrelated to the input), the as-found tolerance has been created. The amount of drift applies only to that which can occur between successive periodic calibrations. The as-found tolerance is the bounding tolerance allowed between calibrations of an instrument or instrument loop. The as-found tolerance establishes the required limits of performance on the device. Any device whose error exceeds the as-found tolerance may be inoperable.

The calculated As-Found Tolerances for the SR and IR channels are as follows:

SR ± 2.25% span
IR ± 1.82% span

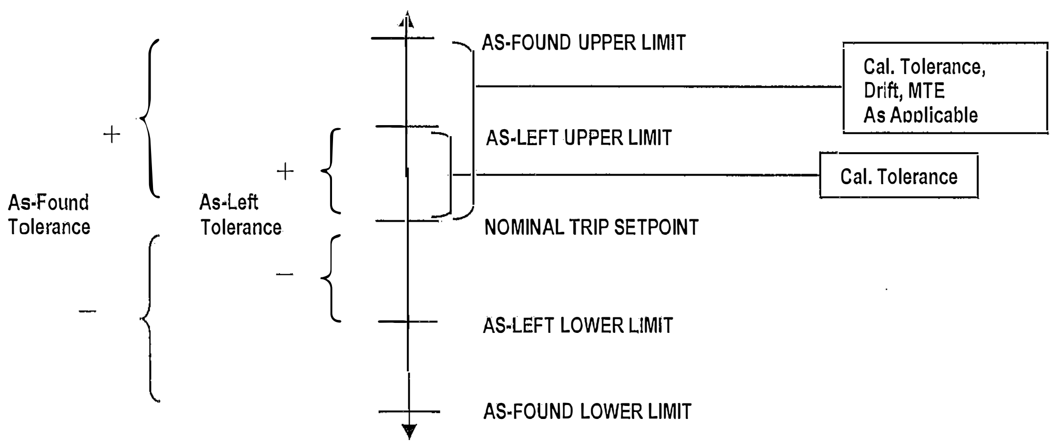
As-Left Tolerance

The as-left tolerance establishes the required accuracy band within which the device or loop segment must be calibrated. The as-left tolerance provides calibration personnel with a measurable calibration band around the nominal Trip Setpoint, within which the device must be adjusted. If the instrument channel setpoint is not reset to a value within the as-left tolerance at the completion of the surveillance, the channel is declared inoperable.

The calculated As-Left Tolerances for the SR and IR channels are as follows:

SR ± 2.02% span
IR ± 1.52% span

Tolerance Relationships



Summary of Calculation Assessment Results

The proposed changes do not affect any safety analysis conclusions because the SR and IR neutron flux reactor trips are not explicitly credited in any safety analyses. The SR and IR neutron flux reactor trips provide redundant protection to the PR neutron flux - low setting reactor trip. The proposed changes to the SR and IR neutron flux reactor trip AVs and permissive P-6 reset will implement values based on the design capabilities of the Thermo Scientific equipment.

3. Changes Related to TSTF-493

The scope of the proposed changes includes two new footnotes that apply to the Channel Functional Test and Channel Calibration for Functional Units 5 and 6 listed in TS 3/4.3.1, Table 4.3-1. The proposed Allowable Value changes affect these Functional Units. The footnotes are consistent with Technical Specification Task Force (TSTF) Change Traveler TSTF-493, Revision 4, “Clarify Application of Setpoint Methodology for LSSS Functions.” The first footnote requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance, but conservative with respect to the AV. The channel evaluation verifies that channel performance continues to satisfy safety analysis assumptions and channel performance assumptions within the setpoint methodology. The purpose of the assessment is to ensure confidence in channel performance prior to returning the channel to service.

The second footnote requires that the as-left setting for the channel be returned to within the as-left tolerance of the nominal Trip Setpoint. Where a setpoint more conservative than the nominal Trip Setpoint is used in the plant surveillance procedures, the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This ensures that sufficient margin is maintained to the SL and/or AL. If the as-left channel setting cannot be returned to within the as-left tolerance of the nominal Trip Setpoint, then the channel shall be declared inoperable. This footnote also indicates that the methodologies used for calculating the as-found and as-left tolerances are specified in the TS Bases.

These footnotes enhance plant safety by ensuring that unexpected as-found conditions are evaluated prior to returning the channel to service, and that as-left settings provide sufficient margin for uncertainties.

4. Technical Analysis Summary

The new Thermo Scientific equipment is equivalent to the existing Westinghouse SR and IR instrumentation with respect to interface with the rest of the NIS and RTS, and meets all the functional requirements of the Westinghouse equipment being replaced. The TS changes described above reflect the operational characteristics of the Thermo Scientific equipment, and do not adversely impact the plant safety analyses and consequently plant safety. TSTF-493 has been appropriately addressed for the proposed changes by adding two notes to the affected Functional Units and specifying the methodologies used for calculating the as-found and as-left tolerances in the TS Bases, which are controlled by 10 CFR 50.59.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment to Renewed Facility Operating License Nos. DPR-70 and 75 for Salem Nuclear Generating Station Units 1 and 2.

The proposed changes to Technical Specification (TS) 3/4.3.1 "Reactor Trip Instrumentation" are needed to support planned plant modifications to replace the existing source range (SR) and intermediate range (IR) nuclear instrumentation. The specific changes include the SR and IR neutron flux reactor trips Allowable Values, and the permissive P-6 reset value. PSEG has evaluated whether or not 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. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The Nuclear Instrumentation System (NIS) provides indication and plant protection through the reactor trip function; it is not an accident initiator or precursor. The reactor trip is part of the plant's accident mitigation response. Thus, the probability of an accident previously evaluated is not significantly increased.

   The performance of the replacement SR and IR detectors and associated equipment will equal or exceed that of the existing Westinghouse instrumentation. The proposed changes are based on accepted industry standards and will preserve assumptions in the applicable accident analyses. The proposed changes do not affect the integrity of the fission product barriers utilized for the mitigation of radiological dose consequences as a result of an accident. The proposed changes do not alter any assumptions previously made in the radiological consequences evaluations, nor do they affect mitigation of the radiological consequences of an accident previously evaluated.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The manner in which the Reactor Trip System (RTS) provides plant protection is not changed. The replacement SR and IR detectors and associated equipment do not affect accident initiation sequences or response scenarios as modeled in the safety analyses. The SR and IR detectors and associated equipment are not accident initiators or precursors. The only physical changes to the plant involve the replacement detectors and associated equipment. The replacement SR and IR detectors and associated equipment have been designed to applicable regulatory and industry standards.

   No changes to the overall manner in which the plant is operated are being proposed. Existing accident scenarios remain unchanged and new or different accident scenarios are not created. The types of accident defined in the Updated Final Safety Analysis Report (UFSAR) continue to represent the credible spectrum of events analyzed to determine safe plant operation.

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

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

   Response: No.

   Margin of safety is related to the confidence in the ability of the fission product barriers to perform their intended functions. These barriers include the fuel cladding, the reactor coolant system pressure boundary, and the containment. Neither the modification to replace the SR and IR detectors and associated equipment, nor the proposed Technical Specification changes will impact these barriers. Accident mitigating equipment will not be adversely impacted as a result of the modification. The safety systems credited in the safety analyses continue to remain available to perform their required mitigation functions. The proposed changes do not affect any safety analysis conclusions because the SR and IR neutron flux reactor trips are not explicitly credited in any accident analyses. Their functional capability enhances the overall reliability of the Reactor Protection System.

   Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based upon the above, PSEG Nuclear LLC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, and, accordingly, a finding of no significant hazards consideration is justified.

5.2 <u>Applicable Regulatory Requirements and Criteria</u>

Salem Nuclear Generating Station, Units 1 and 2 were designed in accordance with the Atomic Industrial Forum (AIF) General Design Criteria (GDC). In addition to the AIF GDC, Salem was designed to comply with Public Service Electric and Gas Company's understanding of the intent of the Atomic Energy Commission (AEC) proposed GDC published in July 1967. As documented in Salem UFSAR Section 3.1, the applicable AEC proposed criteria are Criterion 12 and Criterion 14.

Criterion 12 – Instrumentation and Control Systems. Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Criterion 14 – Core Protection Systems. Core protection systems together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Collectively, these criteria require that instrumentation and controls be provided to monitor and control plant variables in process and protection systems for normal operation, anticipated operational occurrences, and accident conditions. Following implementation of the proposed changes, Salem Nuclear Generating Station, Units 1 and 2 will remain in compliance with AEC proposed Criterion 12 and Criterion 14.

In conclusion, based on the considerations discussed above, (1) there is a 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 NRC'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.

5.3 Precedents

The NRC has approved similar license amendment changes as indicated below:

1. Catawba Units 1 and 2, Amendments 258 and 253 dated August 2, 2010 (TAC Nos. ME1747 and ME1748), Accession No. ML101950353

2. McGuire Units 1 and 2, Amendments 257 and 237 dated August 2, 2010 (TAC Nos. ME1749 and ME1750), Accession No. ML101950451

3. Vogtle Units 1 and 2, Amendments 104 and 82 dated January 22, 1999 (TAC Nos. MA3505 AND MA3506), Accession No. ML012390342

4. Oconee Units 1, 2 and 3, Amendments 223, 223 and 220 dated March 31, 1997 (TAC Nos. M97921, M97922 and M97923), Accession No. ML012000390

6.0 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 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 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.

7.0 REFERENCES

1. NRC Regulatory Guide 1.105, Revision 2, “Instrument Setpoints for Safety-Related Systems,” February, 1986
2. ISA-S67.04-1982, “Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants”
3. WCAP-12103, “Westinghouse Setpoint Methodology for Protection Systems - Salem Units 1 & 2,” (PSEG Identifier S-C-RCP-CDC-0440), October 1989
4. WCAP-14038, “Bases Document for Westinghouse Setpoint Methodology for Protection Systems – Salem Units 1 & 2,” (PSEG Identifier PSBP 320216), December 1994
5. PSEG Nuclear LLC, Technical Standard SC.DE-TS.ZZ-1904(Q), Revision 1, “Instrument Setpoint Calculations for Salem Generating Station Units 1 and 2,” November 2, 2006

TECHNICAL SPECIFICATION PROPOSED CHANGES

The following Technical Specifications for Renewed Facility Operating License DPR-70 are affected by this change request:

Technical Specification	**Page**
Bases Index	XII
3/4.3.1, Table 2.2-1	2-5
3/4.3.1, Table 3.3-1	3/4 3-2
3/4.3.1, Table 3.3-1 (continued)	3/4 3-5
3/4.3.1, Table 3.3-1 (continued)	3/4 3-7
3/4.3.1, Table 4.3-1	3/4 3-11

The following Technical Specifications for Renewed Facility Operating License DPR-75 are affected by this change request:

Technical Specification	**Page**
Bases Index	XII
3/4.3.1, Table 2.2-1	2-5
3/4.3.1, Table 3.3-1	3/4 3-2
3/4.3.1, Table 3.3-1 (continued)	3/4 3-5
3/4.3.1, Table 3.3-1 (continued)	3/4 3-7
3/4.3.1, Table 4.3-1	3/4 3-11

INDEX

BASES

SECTION		PAGE
3/4.3	INSTRUMENTATION	
3/4.3.1	PROTECTIVE	
	AND	
3/4.3.2	ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION	B 3/4 3-1
3/4.3.3	MONITORING INSTRUMENTATION	B 3/4 3-1~~a~~b
3/4.3.4	TURBINE OVERSPEED PROTECTION	B 3/4 3-4
3/4.4	REACTOR COOLANT SYSTEM	
3/4.4.1	REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	B 3/4 4-1
3/4.4.2	SAFETY VALVES	B 3/4 4-1a
3/4.4.3	RELIEF VALVES	B 3/4 4-1a
3/4.4.4	PRESSURIZER	B 3/4 4-2
3/4.4.5	STEAM GENERATOR (SG) TUBE INTEGRITY	B 3/4 4-2
3/4.4.6	REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-4a
3/4.4.7	DELETED	
3/4.4.8	SPECIFIC ACTIVITY	B 3/4 4-5
3/4.4.9	PRESSURE/TEMPERATURE LIMITS	B 3/4 4-6
3/4.4.10	DELETED	B 3/4 4-17
3/4.4.11	BLANK	B 3/4 4-17
3/4.4.12	REACTOR VESSEL HEAD VENTS	B 3/4 4-17

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds
4. Deleted		
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ ~~30~~ 38.5% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ ~~1.3~~ 1.44 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% of instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

* Design flow is 82,500 gpm per loop.

TABLE 3.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Manual Reactor Trip	2	1	2	1,2 and *	12
2.	Power Range, Neutron Flux	4	2	3	1,2, and 3*	2
3.	Power Range, Neutron Flux High Positive Rate	4	2	3	1,2	2
4.	Deleted					
5.	Intermediate Range, Neutron Flux	2	1	2	1,2 and *	3
6.	Source Range, Neutron Flux					
	A. Startup	2	1	2	2## and *	4
	B. Shutdown	2	0	1	3,4, and 5	5
7.	Overtemperature ΔT	4	2	3	1,2	6
8.	Overpower ΔT	4	2	3	1,2	6
9.	Pressurizer Pressure-Low	4	2	3	1,2	6
10.	Pressurizer Pressure--High	4	2	3	1,2	6

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

~~## High voltage to detector may be de-energized above P-6.~~

If ACTION Statement 1 is entered as a result of Reactor Trip Breaker (RTB) or Reactor Trip Bypass Breakers (RTBB) maintenance testing results exceeding the following acceptance criteria, NRC reporting shall be made within 30 days in accordance with Specification 6.9.2:

1. A RTB or RTBB trip failure during any surveillance test with less than or equal to 300 grams of weight added to the breaker trip bar.

2. A RTB or RTBB time response failure that results in the overall reactor trip system time response exceeding the Technical Specification limit.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to ≤ 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to ≤ 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

TABLE 3.3-1 (Continued)

ACTION 10 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY in the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1, provided the other channel is OPERABLE.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 14 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and be in at least HOT STANDBY within 6 hours. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels < ~~6×10^{-11} amps~~ 4.7×10^{-6}% of RTP.	P-6 prevents or defeats the manual block of source range reactor trip.
P-7	With 2 of 4 Power Range Neutron Flux Channels ≥ 11% of RATED THERMAL POWER or 1 of 2 Turbine steam line input pressure channels ≥ a pressure equivalent to 11% of RATED THERMAL POWER.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump undervoltage and under-frequency, pressurizer low pressure, pressurizer high level, and the opening of more than one reactor coolant pump breaker.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(15)	CHANNEL CALIBRATION(15)	CHANNEL FUNCTIONAL TEST(15)	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	(9)	1, 2, and *
2. Power Range, Neutron Flux		(2), (3) (6)		1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	(6)		1, 2
4. Deleted				
5. Intermediate Range, Neutron Flux		(6) #, ##	S/U(1) #, ##	1, 2 and *
6. Source Range, Neutron Flux	(7)	(6) #, ##	(16) and #, ## S/U(1)	2, 3, 4, 5 and *
7. Overtemperature ΔT				1, 2
8. Overpower ΔT				1, 2
9. Pressurizer Pressure--Low				1, 2
10. Pressurizer Pressure--High				1, 2
11. Pressurizer Water Level--High				1, 2
12. Loss of Flow - Single Loop				1

INSERT 1

INSERT 1

If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the nominal Trip Setpoint at the completion of the surveillance; otherwise the channel shall be declared inoperable. Setpoints more conservative than the nominal Trip Setpoint are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The methodologies used to determine the as-found and as-left tolerances are specified in the Technical Specification Bases.

INDEX

BASES

SECTION		PAGE
3/4.3	**INSTRUMENTATION**	
3/4.3.1 AND 3/4.3.2	PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION	B 3/4 3-1
3/4.3.3	MONITORING INSTRUMENTATION	B 3/4 3-1~~a~~ b
3/4.4	**REACTOR COOLANT SYSTEM**	
3/4.4.1	REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	B 3/4 4-1
3/4.4.2 and 3/4.4.3	SAFETY VALVES	B 3/4 4-2
3/4.4.4	PRESSURIZER	B 3/4 4-2
3/4.4.5	RELIEF VALVES	B 3/4 4-2
3/4.4.6	STEAM GENERATOR (SG) TUBE INTEGRITY	B 3/4 4-3
3/4.4.7	REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-4
3/4.4.8	DELETED	
3/4.4.9	SPECIFIC ACTIVITY	B 3/4 4-6
3/4.4.10	PRESSURE/TEMPERATURE LIMITS	B 3/4 4-7
3/4.4.11	DELETED	B 3/4 4-18
3/4.4.12	REACTOR VESSEL HEAD VENTS	B 3/4 4-18

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not applicable	Not applicable
2. Power Range, Neutron Flux	Low setpoint - ≤ 25 % of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 26% of RATED THERMAL POWER High Setpoint - ≤ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 second	≤ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 second
4. Deleted		
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ ~~30~~ 38.5% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ ~~1.3~~ 1.44 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1865 psig	≥ 1855 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2395 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93% instrument span
12. Loss of Flow	≥ 90% of design flow per loop*	≥ 89% of design flow per loop*

*Design flow is 82,500 gpm per loop.

TABLE 3.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2 and *	12
2. Power Range, Neutron Flux	4	2	3	1,2 and 3*	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1,2	2
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1,2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2~~##,~~ and *	4
B. Shutdown	2	0	1	3,4 and 5	5
7. Overtemperature ΔT	4	2	3	1,2	6
8. Overpower ΔT	4	2	3	1,2	6
9. Pressurizer Pressure-Low	4	2	3	1,2	6
10. Pressurizer Pressure--High	4	2	3	1,2	6

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

~~## High voltage to detector may be de-energized above P-6~~.

If ACTION Statement 1 is entered as a result of Reactor Trip Breaker (RTB) or Reactor Trip Bypass Breaker (RTBB) maintenance testing results exceeding the following acceptance criteria, NRC reporting shall be made within 30 days in accordance with Specification 6.9.2:

1. A RTB or RTBB trip failure during any surveillance test with less than or equal to 300 grams of weight added to the breaker trip bar.
2. A RTB or RTBB time response failure that results in the overall reactor trip system time response exceeding the Technical Specification limit.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to ≤ 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to ≤ 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

d. The QUADRANT POWER TILT RATIO, as indicated by the remaining three detectors, is verified consistent with the normalized symmetric power distribution obtained by using either the movable in-core detectors in the four pairs of symmetric thimble locations or the power distribution monitoring system at least once per 12 hours when THERMAL POWER is greater than 75% of RATED THERMAL POWER.

ACTION 10 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY in the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 14 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and be in at least HOT STANDBY within 6 hours. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels < ~~6×10^{-11} amps~~. 4.7×10^{-6}% of RTP	P-6 prevents or defeats the manual block of source range reactor trip.
P-7	With 2 of 4 Power Range Neutron Flux Channels ≥ 11% of RATED THERMAL POWER or 1 of 2 Turbine steam line inlet pressure channels ≥ a pressure equivalent to 11% of RATED THERMAL POWER.	P-7 prevents or defeats the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump undervoltage and under-frequency, pressurizer low pressure, pressurizer high level, and the opening of more than one reactor coolant pump breaker.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK[15]	CHANNEL CALIBRATION[15]	CHANNEL FUNCTIONAL TEST[15]	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	(9)	1, 2, and *
2. Power Range, Neutron Flux		(2), (3) (6)		1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	(6)		1, 2
4. Deleted				
5. Intermediate Range, Neutron Flux		(6) #, ##	S/U[1] #, ##	1, 2 and *
6. Source Range, Neutron Flux	(7)	(6) #, ##	(16) and ~~S/U[1]~~ S/U[1] #, ##	2, 3, 4, 5 and *
7. Overtemperature ΔT				1, 2
8. Overpower ΔT				1, 2
9. Pressurizer Pressure--Low				1, 2
10. Pressurizer Pressure--High				1, 2
11. Pressurizer Water Level--High				1, 2
12. Loss of Flow - Single Loop				1

INSERT 1

INSERT 1

If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the nominal Trip Setpoint at the completion of the surveillance; otherwise the channel shall be declared inoperable. Setpoints more conservative than the nominal Trip Setpoint are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The methodologies used to determine the as-found and as-left tolerances are specified in the Technical Specification Bases.

TECHNICAL SPECIFICATION BASES PROPOSED CHANGES (FOR INFORMATION ONLY)

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

INSERT 1

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests are sufficient to demonstrate this capability. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. Specified surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection

INSERT 1

Two footnotes are added to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for Functional Units 5 and 6 of Table 4.3-1. These footnotes are consistent with Technical Specification Task Force (TSTF) Change Traveler TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions." The first footnote requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance, but conservative with respect to the Allowable Value. The channel evaluation verifies that channel performance continues to satisfy safety analysis assumptions and channel performance assumptions within the setpoint methodology. The purpose of the assessment is to ensure confidence in channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second footnote requires that the as-left setting for the channel be returned to within the as-left tolerance of the nominal Trip Setpoint. This ensures that sufficient margin is maintained to the safety limit and/or analytical limit. If the as-left channel setting cannot be returned to within the as-left tolerance of the nominal Trip Setpoint, then the channel shall be declared inoperable. The as-found tolerance for this function is calculated using the square root sum of the squares combination of uncertainty terms (rack calibration accuracy, rack measurement and test equipment accuracy, rack comparator setting accuracy, and rack drift). The as-left tolerance for this function is calculated using the square root sum of the squares combination of uncertainty terms (rack calibration accuracy, rack measurement and test equipment accuracy, and rack comparator setting accuracy).

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

INSERT 1

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests are sufficient to demonstrate this capability. The Surveillance Frequency is

INSERT 1

Two footnotes are added to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for Functional Units 5 and 6 of Table 4.3-1. These footnotes are consistent with Technical Specification Task Force (TSTF) Change Traveler TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions." The first footnote requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance, but conservative with respect to the Allowable Value. The channel evaluation verifies that channel performance continues to satisfy safety analysis assumptions and channel performance assumptions within the setpoint methodology. The purpose of the assessment is to ensure confidence in channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second footnote requires that the as-left setting for the channel be returned to within the as-left tolerance of the nominal Trip Setpoint. This ensures that sufficient margin is maintained to the safety limit and/or analytical limit. If the as-left channel setting cannot be returned to within the as-left tolerance of the nominal Trip Setpoint, then the channel shall be declared inoperable. The as-found tolerance for this function is calculated using the square root sum of the squares combination of uncertainty terms (rack calibration accuracy, rack measurement and test equipment accuracy, rack comparator setting accuracy, and rack drift). The as-left tolerance for this function is calculated using the square root sum of the squares combination of uncertainty terms (rack calibration accuracy, rack measurement and test equipment accuracy, and rack comparator setting accuracy).