Public Service
Electric and Gas
Company

**Joseph J. Hagan**
Vice President - Nuclear Operations

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NLR-N94137
LCR 94-18

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

REQUEST FOR AMENDMENT
STEAM GENERATOR LOW AND LOW-LOW LEVEL SETPOINTS
SALEM GENERATING STATION UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

In accordance with the requirements of 10CFR50.90, Public Service Electric and Gas Company (PSE&G) hereby transmits a request for amendment of Facility Operating Licenses DPR-70 and DPR-75 for Salem Unit Nos. 1 and 2. Pursuant to the requirements of 10CFR50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

This request would reduce the minimum setpoints and allowable values for the Steam Generator Level-- Low-Low and Low reactor protection system signals. The changes would increase operating margin and reduce the potential for unnecessary reactor trips based on improved steam generator level channel accuracy. This request for amendment satisfies a corrective action identified in Licensee Event Report (LER) 311/94-008-00, dated July 27, 1994, which reported a reactor trip on Steam Generator Level-- Low-Low during plant startup. Changes to the Technical Specification bases are also included to modify and expand the description of the relationship between setpoints, allowable values and the plant safety analyses, based on the improved Standard Technical Specifications of NUREG-1431.

Attachment 1 includes the description and justification for the proposed changes, including PSE&G's Determination of No Significant Hazards Consideration. Attachment 2 contains the Technical Specification pages revised with pen and ink changes.

PSE&G requests an amendment to be implemented at each Salem unit no later than upon restart from the first outage of sufficient duration following issuance of the amendment. These

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implementation provisions would allow adjustment of the setpoints, procedure changes and testing. Approval is requested to allow implementation at Unit 2 during its eighth refueling outage, which is scheduled to begin in October, 1994.

Sincerely,

J. J. Hagan
Vice President -
Nuclear Operations

Affidavit
Attachments (2)

C Mr. T. T. Martin, Administrator - Region I
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REF: NLR-N94137

STATE OF NEW JERSEY )
) SS.
COUNTY OF SALEM )

J. J. Hagan, being duly sworn according to law deposes and says:

I am Vice President - Nuclear Operations of Public Service Electric and Gas Company, and as such, I find the matters set forth in the above referenced letter, concerning the Salem Generating Station, Unit Nos. 1 and 2, are true to the best of my knowledge, information and belief.

JJ Hagan

Subscribed and Sworn to before me
this 19th day of August, 1994

Kimberly Jo Brown
Notary Public of New Jersey

KIMBERLY JO BROWN
NOTARY PUBLIC OF NEW JERSEY
My Commission expires on My Commission Expires April 21, 1998

**ATTACHMENT 1**

## I. DESCRIPTION OF THE PROPOSED CHANGES

Revise Salem Generating Station (SGS) Unit Nos. 1 and 2 Technical Specification (TS) Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, and Table 3.3-4, Engineered Safety Feature Actuation System Instrumentation Trip Setpoints, as follows:

1) Change the Steam Generator Water Level-- Low-Low setpoint from ≥16% Narrow Range Span (NRS) to ≥9.0% NRS, and the allowable value from ≥14.8% NRS to ≥8.0% NRS. This signal is used for reactor trip (Table 2.2-1) and Auxiliary Feedwater (AFW) system actuation (Table 3.3-4).

2) Change the Low Steam Generator Water Level setpoint from ≥25% NRS to ≥10.0% NRS and the allowable value from ≥24% NRS to ≥9.0% NRS. This change would affect the reactor trip functional unit for Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level (Table 2.2-1).

In addition to the proposed Technical Specification changes, revisions to the Technical Specification Bases B2.2.1, Reactor Trip System Instrumentation Setpoints and B3/4.3.1 and 3/4.3.2, Protective and Engineered Safety Features (ESF) Instrumentation are included to clarify the general relationship between setpoints, allowable values and analytical limits used in the safety analyses. A change to the bases for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip is also included (B2.2.1), to delete specific setpoint values.

## II. REASON FOR THE PROPOSED CHANGES

The proposed changes would increase operating margin relative to steam generator level. This would help preclude unnecessary reactor trips and AFW system actuations during plant evolutions involving steam generator water level changes (e.g., plant startup), while continuing to ensure the analytical limits in the safety analyses remain valid.

License Change Request (LCR 92-14), originally submitted via PSE&G letter dated 2/5/93 (NLR-N93001), requested deletion of the reactor trip on Steam/Feedwater Flow Mismatch coincident with Low Steam Generator Level. NRC has reviewed the LCR, as documented in its draft Safety Evaluation Report (SER) letter dated December 2, 1993. Issuance of the NRC amendment is on hold pending PSE&G's replacement of the existing feedwater control system with the Westinghouse advanced digital feedwater control system. The reduction in the steam generator low level setpoint

and allowable value proposed herein would increase operating margin associated with the trip function until removal of the trip function is implemented.

The Bases changes were developed using the improved Standard Technical Specification bases for Trip Setpoints and Allowable Values (NUREG-1431, B3.3.1 and B3.3.2). These changes would incorporate a description of the relationship between setpoints, allowable values and analytical limits, which are determined by PSE&G's setpoint methodology consistent with standard industry practice.

## III. **JUSTIFICATION FOR THE PROPOSED CHANGES**

The proposed changes are based on reduced channel uncertainties that have been calculated by PSE&G using a setpoint methodology consistent with Instrument Society of America standard ISA-S67.04, which is endorsed by NRC Regulatory Guide 1.105, Rev. 2. The reduction in channel uncertainty is primarily the result of replacing the Rosemount 1153 series level transmitters with Rosemount 1154HH transmitters, which have improved environmental specifications. The total accident channel uncertainty previously prepared utilizing the Rosemount 1153 transmitters resulted in a 15.3% NRS error. This total accident uncertainty as calculated by PSE&G utilizing the Rosemount 1154HH transmitters has been reduced to 7.407% NRS. The reduction of total accident uncertainties included a more specific analysis of process measurement uncertainties resulting in an improved overall value. Based on the reduced uncertainties, the proposed setpoints and allowable values would continue to ensure the trip settings assumed in the plant safety analyses remain valid.

1) The **Steam Generator Water Level-- Low-Low** signal initiates a reactor trip and actuation of the Auxiliary Feedwater (AFW) system. This signal is used as a primary protection signal for postulated design basis events including loss of normal feedwater, loss of offsite power and feedwater line break. The safety analyses assume reactor trip and AFW actuation occurs at 0.0% NRS (i.e., analytical limit). The total calculated channel uncertainty for the low-low level channel is +7.407%, -3.458%. Because the low-low level signal protects against conditions involving decreasing steam generator level, the positive uncertainty value (+7.407%) is subtracted from the setpoint to determine whether there is adequate margin relative to the analytical limit. The proposed setpoint of ≥9.0% and allowable value of ≥8.0% would ensure the analytical limit of 0.0% NRS is met with excess margin.

2) The **Low Steam Generator Water Level** signal coincident with the Steam Flow/Feed Flow Mismatch signal initiates a reactor trip. This signal is not credited in any safety analyses, but increases the overall reliability of the reactor protection system. The low-low level signal uses three channels per steam generator, one of which is used by the

level control system. The low level coincident trip is designed to ensure compliance with IEEE-279-1971, Section 4.7.3 relative to control and protection system interactions, such that a control system failure and random single failure of a protection channel would not result in loss of the protection function.

Because it is not credited in the UFSAR Chapter 15 safety analyses, there is no analytical limit associated with the low level signal. The uncertainties calculated for the low level signal are identical to that of the low-low level signal (+7.407%, -3.458%). The proposed setpoint of ≥10.0% NRS and allowable value of ≥9.0% NRS would continue to provide backup protection to the low-low level trip signal with excess margin to the analytical limit used for the low-low level trip. The reduction in setpoint would increase the margin available for steam generator level recovery when a flow mismatch condition exists.

The bases changes for the low level coincident trip would delete the specific setpoints for the flow mismatch and low steam generator level signals. Specific setpoint values are maintained in the Technical Specifications, and need not be duplicated in the bases. The overall intent of the trip function is not changed. As stated in the current bases, the trip would continue to be initiated before the steam generators are dry, reducing demands on the AFW system and minimizing thermal transients. Note that the AFW system is actuated by the Low-Low Level signal (not the low level coincident signal). The analytical limit for the low-low setpoint (0.0% NRS) is above the top of the steam generator tubes.

The accompanying bases changes relative to setpoints and allowable values are based on the improved Westinghouse Standard Technical Specifications (NUREG-1431 B3.3.1). Significant differences between the NUREG-1431 bases and the changes included herein are as follows:

- NUREG-1431 refers to "RTS/ESFAS Setpoint Methodology Study" where the Salem bases refer to ISA-S67.04-1982. This is because the Salem setpoint methodology is consistent with the ISA standard, which is widely used in the industry and endorsed by NRC in Regulatory Guide 1.105, Rev. 2. PSE&G does not use a document called "RTS/ESFAS Setpoint Methodology Study," and believes it is more appropriate to refer to an industry standard than a plant-specific study in the Technical Specification bases.

- NUREG-1431 refers to the CHANNEL OPERATIONAL TEST (COT) as the test which is capable of detecting those measurement uncertainties comprising the difference between the Trip Setpoint and Allowable Value. The Salem bases refer to the CHANNEL FUNCTIONAL TEST, which is equivalent to the COT in NUREG-1431. CHANNEL FUNCTIONAL TEST is a defined term in

the Salem Technical Specifications, whereas COT is not.

- NUREG-1431 includes a paragraph relative to the ability to test channels on-line "to verify that the signal or setpoint accuracy is within the specified allowance requirements of [UFSAR Chapter 6]..." This paragraph is not included in the Salem bases. Salem UFSAR Chapter 6 does not specify channel "allowance requirements." The Salem Technical Specifications and bases already define test requirements in sufficient detail such that the paragraph in NUREG-1431 is not considered necessary.

The bases changes are not unique to the steam generator level setpoints and allowable values proposed herein, but are included because they are useful in describing the method in which setpoints and allowable values are established, consistent with an approved industry standard, to ensure that the reactor protection system channels protect the limits of the safety analyses.

## IV. **DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION**

The proposed changes for Salem Unit Nos. 1 and 2:

(1) do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Steam Generator Water Level-- Low-Low signal and the Low Steam Generator Level coincident with Steam Flow/Feed Flow Mismatch signal are designed to mitigate design basis transients involving significant reductions of steam generator inventory (e.g., Loss of Normal Feedwater, Turbine Trip, Loss of Offsite Power, Feedwater Line Break). The setpoints and allowable values for these protection signals are prescribed by Technical Specifications such that performance of the signals is consistent with the plant safety analyses, considering the effects of channel uncertainties. The proposed reductions to the setpoints and allowable values for the low-low and low steam generator level signals would not affect the probability of any transient that the protection signals are designed to mitigate. The changes would reduce the probability of unnecessary reactor trips and Auxiliary Feedwater (AFW) system actuations by providing greater operating margin for plant evolutions involving steam generator level changes (e.g., plant startup). Therefore, the proposed changes do not involve any increase in probability of an accident previously evaluated.

The changes to the Steam Generator Water Level-- Low-Low signal would not result in any increase in consequences of a previously analyzed accident because the proposed setpoint and allowable value would continue to ensure the safety analysis assumptions remain valid. As described in the accompanying changes to the Technical Specification Bases, the channel uncertainty calculations performed to establish the relationships between the setpoints, allowable values and safety analyses are consistent

with NRC Regulatory Guide 1.105, Revision 2. Low Steam Generator Level coincident with Steam Flow/Feed Flow Mismatch signal is not credited in the UFSAR Chapter 15 safety analyses. The proposed changes to the low steam generator level setpoint and allowable value would continue to provide reliable backup to the low-low level trip signal, consistent with IEEE-279-1971. Therefore, the proposed changes would not involve an increase in consequences of any previously analyzed accident.

(2) do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes would continue to ensure the appropriate reactor protection system functions (reactor trip and AFW initiation) are initiated in the event that steam generator water level decreases to the value used in the plant safety analyses. The proposed changes would not involve any changes in protection system logic or function, and do not involve any plant configurations that could adversely affect the initiation or progression of any accident sequence. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) do not involve a significant reduction in a margin of safety.

The proposed setpoints and allowable values would continue to ensure that the assumptions in the safety analyses remain valid, with appropriate consideration of protection system channel uncertainties. Therefore, the proposed changes do not involve a reduction in margin of safety.

Therefore, PSE&G has concluded that the changes proposed herein do not involve a Significant Hazards Consideration.

ATTACHMENT 2

## TABLE 2.2-1 (Continued)

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 13. Steam Generator Water Level--Low-Low | ≥ ~~16%~~ 9.0% of narrow range instrument span-each steam generator | ≥ ~~14.8%~~ 8.0% of narrow range instrument span-each steam generator |
| 14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | ≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ ~~25%~~ 10.0% of narrow range instrument span--each steam generator | ≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ ~~24%~~ 9.0% of narrow range instrument span--each steam generator |
| 15. Undervoltage-Reactor Coolant Pumps | ≥ 2900 volts-each bus | ≥ 2850 volts-each bus |
| 16. Underfrequency-Reactor Coolant Pumps | ≥ 56.5 Hz - each bus | ≥ 56.4 Hz - each bus |
| 17. Turbine Trip | | |
| A. Low Trip System Pressure | ≥ 45 psig | ≥ 45 psig |
| B. Turbine Stop Valve Closure | ≤ 15% off full open | ≤ 15% off full open |
| 18. Safety Injection Input from SSPS | Not Applicable | Not Applicable |
| 19. Reactor Coolant Pump Breaker Position Trip | Not Applicable | Not Applicable |

reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by $\geq 1.42 \times 10^6$ lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

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Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### BASES

### 2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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~~The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.~~

#### Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

#### Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 9 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

#### Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

BASES

through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature $\Delta T$ trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature $\Delta T$ trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients with 3 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall

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## TABLE 3.3-4 (Continued)

## ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 5. TURBINE TRIP AND FEEDWATER ISOLATION | | |
| A. Steam Generator Water Level -- High-High | ≤ 67% of narrow range instrument span each steam generator | ≤ 68% of narrow range instrument span each steam generator |
| 6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC) | Not Applicable | Not Applicable |
| 7. UNDERVOLTAGE, VITAL BUS | | |
| a. Loss of Voltage | ≥ 70% of bus voltage | ≥ 65% of bus voltage |
| b. Sustained Degraded Voltage | ≥ 91.6% of bus voltage for ≤ 13 seconds | ≥ 91% of bus voltage for ≤ 15 seconds |
| 8. AUXILIARY FEEDWATER | | |
| a. Automatic Actuation Logic | Not Applicable | Not Applicable |
| b. Manual Initiation | Not Applicable | Not Applicable |
| c. Steam Generator Water Level--Low-Low | ≥ ~~16%~~ 9.0% of narrow range instrument span each steam generator | ≥ ~~14.8%~~ 8.0% of narrow range instrument span each steam generator |
| d. Undervoltage - RCP | ≥ 70% RCP bus voltage | ≥ 65% RCP bus voltage |
| e. S.I. | See 1 above (All S.I. setpoints) | |
| f. Trip of Main Feedwater Pumps | Not Applicable | Not Applicable |
| g. Station Blackout | See 6 and 7 above (SEC and Undervoltage, Vital Bus) | |

## 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.

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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.

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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 performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and 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 Instrumentation System," and Supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

## 3/4.3.3 MONITORING INSTRUMENTATION

### 3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served

## TABLE 2.2-1 (Continued)

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
| --- | --- | --- |
| 13. Steam Generator Water Level--Low-Low | ≥ ~~16%~~ 9.0% of narrow range instrument span-each steam generator | ≥ ~~14.8%~~ 8.0% of narrow range instrument span-each steam generator |
| 14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | ≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ ~~25%~~ 10.0% of narrow range instrument span--each steam generator | ≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ ~~24%~~ 9.0% of narrow range instrument span--each steam generator |
| 15. Undervoltage-Reactor Coolant Pumps | ≥ 2900 volts-each bus | ≥ 2850 volts-each bus |
| 16. Underfrequency-Reactor Coolant Pumps | ≥ 56.5 Hz - each bus | ≥ 56.4 Hz - each bus |
| 17. Turbine Trip | | |
| A. Low Trip System Pressure | ≥ 45 psig | ≥ 45 psig |
| B. Turbine Stop Valve Closure | ≤ 15% off full open | ≤ 15% off full open |
| 18. Safety Injection Input from SSPS | Not Applicable | Not Applicable |
| 19. Reactor Coolant Pump Breaker Position Trip | Not Applicable | Not Applicable |

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

## BASES

### 2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

REPLACE WITH INSERT 1

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineering Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

#### Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

#### Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above approximately 9 percent of RATED THERMAL POWER) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POWER).

#### Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature delta T trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients with 3 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. ~~The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to $1.42 \times 10^6$ lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 24 percent, as indicated by the narrow range instrument. These~~ trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

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## ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 7. UNDERVOLTAGE, VITAL BUS | | |
| a. Loss of Voltage | ≥ 70% of bus voltage | ≥ 65% of bus voltage |
| b. Sustained Degraded Voltage | ≥ 91.6% of bus voltage for ≤ 13 seconds | ≥ 91% of bus voltage for ≤ 15 seconds |
| 8. AUXILIARY FEEDWATER | | |
| a. Automatic Actuation Logic | Not Applicable | Not Applicable |
| b. Manual Initiation | Not Applicable | Not Applicable |
| c. Steam Generator Water Level--Low-Low | ≥ ~~16%~~ 9.0% of narrow range instrument span each steam generator | ≥ ~~14.8 %~~ 8.0% of narrow range instrument span each steam generator |
| d. Undervoltage - RCP | ≥ 70% RCP bus voltage | ≥ 65% RCP bus voltage |
| e. S.I. | See 1 above (all S.I. setpoints) | |
| f. Trip of Main Feedwater Pump | Not Applicable | Not Applicable |
| g. Station Blackout | See 6 and 7 above (SEC and Undervoltage, Vital Bus) | |
| 9. SEMIAUTOMATIC TRANSFER TO RECIRCULATION | | |
| a. RWST Low Level | 15.25 ft. above Instrument taps | 15.25 ± 1 ft. above instrument taps |
| b. Automatic Actuation Logic | Not Applicable | Not Applicable |

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## 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 ~~functions~~ capability is available from diverse parameters.
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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.

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 performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and 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 Instrumentation System," and Supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

## 3/4.3.3 MONITORING INSTRUMENTATION

### 3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

Insert 1

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.