Mr. Leon R. Eliason
Chief Nuclear Officer and President-
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

November 4, 1994

SUBJECT: REVISE STEAM GENERATOR WATER LOW-LOW AND LOW LEVEL TRIP SETPOINTS, SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 (TAC NOS. M90215 AND M90216)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 159 and 140 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 19, 1994, and supplemented October 4, 1994.

These amendments reduce the minimum setpoints and allowable values for the steam generator low and low-low level reactor protection system signals.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
/S/
Leonard N. Olshan, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-272/50-311

Enclosures:
1. Amendment No. 159 to License No. DPR-70
2. Amendment No. 140 to License No. DPR-75
3. Safety Evaluation

cc w/encls:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 4, 1994

Mr. Leon R. Eliason
Chief Nuclear Officer and President-
Nuclear Business Unit
Public Service Electric & Gas
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Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: REVISE STEAM GENERATOR WATER LOW-LOW AND LOW LEVEL TRIP SETPOINTS, SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 (TAC NOS. M90215 AND M90216)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 159 and 140 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 19, 1994, and supplemented October 4, 1994.

These amendments reduce the minimum setpoints and allowable values for the steam generator low and low-low level reactor protection system signals.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

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Leonard N. Olshan, Senior Project Manager
Project Directorate I-2
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Docket Nos. 50-272/50-311

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3. Safety Evaluation

cc w/encls:
See next page

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PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 159
License No. DPR-70

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated August 19, 1994, as supplemented October 4, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 159 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented at the next outage of sufficient duration.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 4, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 159

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-6	2-6
B 2-3	B 2-3
B 2-4	B 2[illegible]
B 2-7	B 2-7
3/4 3-26	3/4 3-26
B [illegible]/4 3-1	B 3/4 3-1
-	B 3/4 3-1a

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 9.0% of narrow range instrument span-each steam generator	≥ 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 ≥ 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 ≥ 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

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.

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

LIMITING SAFETY SYSTEM SETTINGS

BASES

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.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR value for multiple control rod drop accidents. The analysis of a single rod drop accident indicates a return to full power may be initiated by the automatic control system in response to a continued full power turbine load demand or by the negative moderator temperature feedback. This transient will not result in a DNBR of less than the design DNBR value, therefore single rod drop protection is not required.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Δ T

The Overtemperature Δ T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The 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.

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.

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	≥ 9.0% of narrow range instrument span each steam generator	≥ 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.

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.

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated August 19, 199[illegible], as supplemented October 4, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 140 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented during the eighth refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 4, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 140

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-6	2-6
B 2-3	B 2-3
B 2-4	B 2-4
B 2-6	B 2-6
3/4 3-27	3/4 3-27
B 3/4 3-1	B 3/4 3-1
-	B 3/4 3-1a

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 9.0% of narrow range instrument span-each steam generator	≥ 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 ≥ 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 ≥ 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

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.

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 autom...ically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below approximately 9 percent of RATED THERMAL POW...R).

Power Range, Neut...n 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.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR value for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which, when in conju... ion with nuclear power being maintai... equivalent to turbine pow... by action of the automatic rod control system, ...ld cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor for all single or multiple dropped rods.

Intermediate and Source Range, Nuclear Fl

The Intermedi...e and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup. The trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 s...onds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peak... are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

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 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.

TABLE 3.3-4

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	≥ 9.0% of narrow range instrument span each steam generator	≥ 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 P [illegible]ECTIVE AND ENGINEERED SAFETY FEATURES (E[illegible]) 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 s[illegible]point, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenan[illegible] 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 ass[illegible] available in t[illegible] 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 Set[illegible]ts 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.

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 159 AND 140 TO FACILITY OPERATING

LICEN NOS. DPR-70 AND DPR-75

PUBLIC SERVICE ELECTRIC & G COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated August 19, 1994, as supplemented October 4, 19 , Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit Nos. 1 and 2, Technical Specifications (TS). The requested changes would revise the steam genera water low-low and low level trip setpoints. The changes would increase the operating margin relative to steam generator level which would help preclude unnecessary reactor trips and auxiliary feedwater system (AFW) actuations during plant evolutions involving steam generator water level changes. The supplemental letter provides additional information, but does not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The proposed changes are based on reduced channel uncertainties that have been calculated by the licensee using a setpoint methodology consistent with the Instrument Society of America (ISA) Standard, S67.04-1982, "Setpoints for Nuclear Safety Related Instrumentation," which is endorsed by Regulatory Guide (RG) 1.105, Revision (Rev.) 2, "Instrumentation Setpoints for Nuclear Safety Related Instrumentation." The reduction in channel uncertainty is primarily the result of replacing the Rosemount 1153 series level transmitters with Rosemount 1154HH transmitters.

The total accident uncertainty with the Rosemount 1153 series transmitters resulted in a 15.3% narrow range span (NRS) error. This accident uncertainty with the Rosemount 1154HH tran tters was reduced to 7.407% NRS. Based on t reduced uncertainties, the setpoints and allowable values can be reduced and still ensure the analytical limit of 0.0% NRS is met with excess margin.

The steam generator water low-low level signal initiates a reactor trip and actuation of the 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 proposed changes would revise the steam generator water low-low l l setpoint, in TS Tables 2.2-1 and 3.3-4, from ≥16% NRS to ≥9.0% NRS, and   allowable value from ≥14.8% NRS to ≥8.0% NRS. The proposed setpoint and allowable value would ensure the analytical limit of 0.0% NRS is met with excess margin. The proposed reductions to the setpoints and allowable values for the l -low and low steam generator le   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 actuation by providing greater op ating rgin for plant evolutions involving steam generator changes (e.g., plant startup). Therefore, the   posed changes do not involve any increase in the probability of an accident previously evaluated.

The steam generator water low level signal coincident with the steam flow/feed flow mi tch signal initiates a reactor trip. This signal is not credited in the safety analysis, but increases the overall reliability of the RPS. The proposed changes would revise the steam generator water low level and steam/feedwater flow mismatch setpoint, in TS Table 2.2-1, from ≥25% NRS to ≥10.0% NRS, and the allowable value from ≥24% NRS to ≥9.0% NRS. Because it is not credited in the safety analysis there is no analytical limit associated with the steam generator water low level signal. The uncertainties calculated for the steam generator water lo level signal are identical to those of the low-low signal. The reduction in the low level setpoint would increase t margin available for steam generator water level recovery when a flow mismatch condition exists.

The proposed changes to TS Basis 2.2.1, Reactor Trip System Instrume tion Setpoints and TS Bases 3/4.3.1 and 3/4.3.2, Protectiv and Engineering Safety Features Instrumentation are to clarify the general r ationship between setpoints, allowable values, and analytical limits used in the safety analysis. These changes are based on the improved Westinghouse Standard TSs NUREG-1431 Bases 3.3.1. The licensee has provided justification for differences between NUREG-1431 and the proposed changes.

NUREG-1431 refere es "RTS/ESFAS Setpoint Methodology Study." The licensee's setpoint methodology is not based on the "RTS/ESFAS Setpoint Methodology Study", but rather ISA Standard S67.04-1982 which is widely used in industry and is endorsed by RG 1.105, Rev. 2.

NUREG-1431 refers to the CHANNEL OPERATIONAL TEST (COT) as the test which is capable of detecting those measurement uncertainties compri ing the differences between the trip setpoint and the allowable value. The licensee's bases refer to a CHANNEL FUNCTIONAL TE   (CFT), which is equivalent to the COT in NUREG-1431. The CFT is defined and specified in the licensee's TSs.

NUREG-1431 includes a paragraph that discusses 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]..." The licensee's bases do not include a paragraph with this information and UFSAR Chapter 6 does not specify channel "allowance requirements". However, the licensee's TSs and Bases define test requirements in sufficient detail so that a paragraph similar to the paragraph in NUREG-1431 is not necessary.

We have reviewed the licensee's calculation for the steam generator water low-low and low level setpoints and allowable values that were submitted October 4, 1994, and have found them to be consistent with the guidelines specified in ISA Standard S67.04-1982.

Based on the above, the staff finds the proposed changes to the Salem Generating Station Unit Nos. 1 and 2, TS Table 2.2-1, TS Table 3.3-4 and Bases 2.2.1, 3/4.3.1 and 3/4.3.2 acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 47180). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is 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 Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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