January 17, 7

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

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 (TAC NOS. M96665 AND M96666)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 187 and 170 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 October 1, 1996, supplemented October 31, 1996.

These amendments change TS 3/4.7.1.5, "Main Steam Line Isolation Valves (MSIVs)," and 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation." These changes have been made to accommodate entry into Modes 3 and 2 prior to performing MSIV closure time testing in Mode 2. The amendments also allow additional time for the repair and testing of inoperable MSIVs in certain operating Modes, delete footnotes that are no longer applicable, and change the low steam line pressure trip setpoint value for safety injection, turbine trip and feedwater isolation to make it consistent with the actual plant configuration.

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. 50-272/311

Enclosures: 1. Amendment No. 187 to License No. DPR-70
2. Amendment No. 170 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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Docket File	JStolz	GHill(4)	LNicholson, RGN-I
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NAME	LOlshan:rb	MO'Brien	TMarsh	CGrimes		JStolz
DATE	12/12/96	12/X/96	12/16/96	12/26/96	1/17/96	1/15/96

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

January 17, 1997

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

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 (TAC NOS. M96665 AND M96666)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 187 and 170 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 October 1, 1996, supplemented October 31, 1996.

These amendments change TS 3/4.7.1.5, "Main Steam Line Isolation Valves (MSIVs)," and 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation." These changes have been made to accommodate entry into Modes 3 and 2 prior to performing MSIV closure time testing in Mode 2. The amendments also allow additional time for the repair and testing of inoperable MSIVs in certain operating Modes, delete footnotes that are no longer applicable, and change the low steam line pressure trip setpoint value for safety injection, turbine trip and feedwater isolation to make it consistent with the actual plant configuration.

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

Sincerely,

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

Docket Nos. 50-272/311

Enclosures:
1. Amendment No. 187 to License No. DPR-70
2. Amendment No. 170 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

Mr. Leon R. Eliason
Public Service Electric & Gas
Company

Salem Nuclear Generating Station,
Units 1 and 2

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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. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 187
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 October 1, 1996, supplemented October 31,1996, 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.

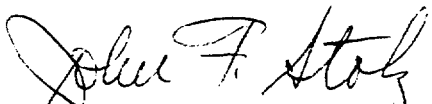
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 187 , 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 prior to entry into Mode 3 from the current outage.

FOR THE NUCLEAR REGULATORY COMMISSION



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: January 17, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 187

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 3-14	3/4 3-14
3/4 3-23	3/4 3-23
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-32a	3/4 3-32a
3/4 3-34	3/4 3-34
3/4 7-10	3/4 7-10

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3. The provisions of Specification 4.0.4 are not applicable to MSIV closure time testing.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	≤4.0 psig	≤4.5 psig
d. Pressurizer Pressure--Low	≥1765 psig	≥1755 psig
e. Differential Pressure Between Steam Lines--High	≤100 psi	≤112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	≤A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load T_{avg} ≥ 543°F ≥ 600 psig steam line pressure	≤A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load T_{avg} ≥ 541°F ≥ 579 psig steam line pressure

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
3.	Pressurizer Pressure-Low	
a.	Safety Injection (ECCS)	≤ 27.0[1]/12.0[2]
b.	Reactor Trip (from SI)	≤ 2.0
c.	Feedwater Isolation	≤ 10.0
d.	Containment Isolation - Phase "A"	≤ 18.0[2]
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	≤ 60
g.	Service Water System	≤ 49.0[1]/13.0[2]
4.	Differential Pressure Between Steam Lines-High	
a.	Safety Injection (ECCS)	≤ 12.0[2]/22.0[3]
b.	Reactor Trip (from SI)	≤ 2.0
c.	Feedwater Isolation	≤ 10.0
d.	Containment Isolation - Phase "A"	≤ 17.0[2]/27.0[3]
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	≤ 60
g.	Service Water System	≤ 13.0[2]/48.0[3]
5.	Steam Flow in Two Steam Lines - High Coincident with T_{avg} -- Low-Low	
a.	Safety Injection (ECCS)	≤ 15.75[2]/25.75[3]
b.	Reactor Trip (from SI)	≤ 5.75
c.	Feedwater Isolation	≤ 15.0
d.	Containment Isolation - Phase "A"	≤ 20.75[2]/30.75[3]
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	≤ 61.75
g.	Service Water System	≤ 15.75[2]/50.75[3]
h.	Steam Line Isolation	≤ 10.75

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
6.	Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a.	Safety Injection (ECCS)	≤ 12.0[2]/22.0[3]
b.	Reactor Trip (from SI)	≤ 2.0
c.	Feedwater Isolation	≤ 10.0
d.	Containment Isolation-Phase "A"	≤ 17.0[2]/27.0[3]
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	≤ 60
g.	Service Water System	≤ 14.0[2]/48.0[3]
h.	Steam Line Isolation	≤ 8.0
7.	Containment Pressure--High-High	
a.	Containment Spray	≤ 33.0
b.	Containment Isolation-Phase "B"	Not Applicable
c.	Steam Line Isolation	≤ 7.0
8.	Steam Generator Water Level--High High	
a.	Turbine Trip	≤ 2.5
b.	Feedwater Isolation	≤ 10.0
9.	Steam Generator Water Level--Low-Low	
a.	Motor-Driven Auxiliary Feedwater Pumps(4)	≤ 60.0
b.	Turbine-Driven Auxiliary Feedwater Pumps(5)	≤ 60.0

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1,2,3**
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3
c. Containment Pressure--High-High	S	R	Q(3)	1,2,3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	S	R	Q	1,2,3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	Q	1,2,3
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC) LOGIC				
a. Inputs	N.A.	N.A.	M(6)	1,2,3,4
b. Logic, Timing and Outputs *	N.A.	N.A.	M(1)	1,2,3,4
7. UNDERVOLTAGE, VITAL BUS				
a. Loss of Voltage	S	R	M	1,2,3
b. Sustained Degraded Voltage	S	R	M	1,2,3

TABLE 4.3-2 (Continued)

TABLE NOTATION

* Outputs are up to, but not including, the output relays.

** The provisions of Specification 4.0.4 are not applicable.

(1) Each logic channel shall be tested at least once per 62 days on a STAGGERED TEST BASIS. The CHANNEL FUNCTION TEST of each logic channel shall verify that its associated diesel generator automatic load sequence timer is OPERABLE with the interval between each load block within 1 second of its design interval.

(2) Each train or logic channel shall be tested at least every 62 days on a staggered basis.

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

(4) NOT USED

(5) NOT USED

(6) Inputs from Undervoltage, Vital Bus, shall be tested monthly. Inputs from Solid State Protection System shall be tested every 62 days on a STAGGERED TEST BASIS.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

otherwise, be in MODE 2 within the next 6 hours.

MODES 2 and 3 - With one or more main steam line isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may proceed provided;

a. The isolation valve(s) is (are) maintained closed, and

b. The isolation valve(s) is (are) verified closed once per 7 days.

Otherwise, be in MODE 3, HOT STANDBY, within the next 6 hours, and MODE 4, HOT SHUTDOWN, within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable.

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. 170
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 October 1, 1996, supplemented October 31, 1996, 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) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 170 , 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 prior to entry into Mode 3 from the current outage.

FOR THE NUCLEAR REGULATORY COMMISSION

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: January 17, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 3-14	3/4 3-14
3/4 3-24	3/4 3-24
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-35	3/4 3-35
3/4 3-37	3/4 3-37
3/4 7-10	3/4 7-10

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3. The provisions of Specification 4.0.4 are not applicable to MSIV closure time testing.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	≤ 4.0 psig	≤ 4.5 psig
d. Pressurizer Pressure--Low	≥ 1765 psig	≥ 1755 psig
e. Differential Pressure Between Steam Lines--High	≤ 100 psi	≤ 112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low or Steam Line Pressure--Low	≤ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load T_{avg} ≥ 543°F ≥ 600 psig steam line pressure	≤ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load T_{avg} ≥ 541°F ≥ 579 psig steam line pressure

TABLE 3.3.5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0[1]/12.0[2]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	≤ 18.0[2]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 49.0[1]/13.0[2]
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤ 12.0[2]/22.0[3]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation Phase "A"	≤ 17.0[2]/27.0[3]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 13.0[2]/48.0[3]
5. Steam Flow in two Steam Lines High-Coincident with T_{avg} --Low-Low	
a. Safety Injection (ECCS)	≤ 15.75[2]/25.75[3]
b. Reactor Trip (from SI)	≤ 5.75
c. Feedwater Isolation	≤ 15.0
d. Containment Isolation-Phase "A"	≤ 20.75[2]/30.75[3]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 61.75
g. Service Water System	≤ 15.75[2]/50.75[3]
h. Steam Line Isolation	≤ 10.75

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 12.0[2]/22.0[3]
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	≤ 17.0[2]/27.0[3]
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	≤ 14.0[2]/48.0[3]
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 33.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 10.0
9. Steam Generator Water Level --Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps(4)	≤ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps(5)	≤ 60.0

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1,2,3**
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3
c. Containment Pressure--High-High	S	R	Q(3)	1,2,3
d. Steam Flow in Two Steam Lines--High Coincident with Tavg--Low-Low or Steam Line Pressure--Low	S	R	Q	1,2,3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	Q	1,2,3
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC) LOGIC				
a. Inputs	N.A.	N.A.	M(6)	1,2,3,4
b. Logic, Timing and Outputs *	N.A.	N.A.	M(1)	1,2,3,4
7. UNDERVOLTAGE, VITAL BUS				
a. Loss of Voltage	S	R	M	1,2,3
b. Sustained Degraded Voltage	S	R	M	1,2,3

TABLE 4.3-2 (Continued)

TABLE NOTATION

* Outputs are up to, but not including, the Output Relays.

** The provisions of Specification of 4.0.4 are not applicable.

(1) Each logic channel shall be tested at least once per 62 days on a STAGGERED TEST BASIS. The CHANNEL FUNCTION TEST of each logic channel shall verify that its associated diesel generator automatic load sequence timer is OPERABLE with the interval between each load block within 1 second of its design interval.

(2) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

(4) If not performed in the previous 92 days.

(5) NOT USED

(6) Inputs from undervoltage, Vital Bus, shall be tested monthly. Inputs from Solid State Protection System, shall be tested every 62 days on a STAGGERED TEST BASIS.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours;

Otherwise, be in MODE 2 within the next 6 hours.

MODES 2 and 3 - With one or more main steam line isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may proceed provided;

a. The isolation valve(s) is (are) maintained closed, and

b. The isolation valve(s) is (are) verified closed once per 7 days.

Otherwise, be in MODE 3, HOT STANDBY, within the next 6 hours, and MODE 4, HOT SHUTDOWN, within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specificaiton 4.0.4 are not applicable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 187 AND 170 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PUBLIC SERVICE ELECTRIC & GAS 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 October 1, 1996, as supplemented October 31, 1996, the 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 (TSs). The requested changes would revise TS 3/4.7.1.5, "Main Steam Line Isolation Valves (MSIVs)," and TS 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation." These changes are needed to accommodate entry into Mode 2. The proposed amendments would allow for the repair and testing of inoperable MSIVs in certain operating Modes, and would change the low steam line pressure trip setpoint value for safety injection to make it consistent with the previously approved value for steam line isolation. The October 31, 1996, letter proposed changes to provide greater consistency with the requirements of NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1. These changes did not change the initial proposed no significant hazards consideration determination or the Federal Register notice.

2.0 EVALUATION

The safety function of the MSIVs is to close automatically in the event of a main steam line break or a malfunction that results in a secondary system depressurization. Because steam pressure of the Main Steam System assists valve closure, the MSIVs must be tested in higher modes in order to meet the 5-second closure time specified in TS 4.7.1.5. The steam pressure on entry into Mode 3 is not sufficient to meet the 5-second closure time. However, TS 4.0.4 prohibits entry into an OPERATIONAL MODE unless the Surveillance Requirement associated with the Limiting Condition for Operation has been performed. Thus, the licensee proposed changes to TS 4.3.2.1.3, TS Table 4.3-2, and TS 4.7.1.5 that make the provisions of TS 4.0.4 not applicable to MSIV closure time testing. The staff agrees this change is necessary to accommodate MSIV closure time testing and therefore finds it acceptable.

The licensee also proposed a change to TS 3.7.1.5 to allow, with one MSIV inoperable in Mode 1, 6 hours to be in Mode 2. The existing TS allowed 12 hours to be in HOT SHUTDOWN, Mode 4, with one MSIV inoperable in Mode 2. Thus, the proposed change would allow a total of 18 hours to be in HOT SHUTDOWN with one inoperable MSIV, an increase of 6 hours over the existing TS. The licensee stated that the 6 hours to be in Mode 2 is considered a reasonable amount of time, based on operating experience, to reach Mode 2 and to close the MSIVs in an orderly manner without challenging plant systems. In response to an NRC concern, the licensee in its October 31, 1996, letter proposed additional changes to provide greater consistency with NUREG-1431. The staff agrees that the times to reach Modes 2, 3 and 4 that were proposed by the licensee are reasonable, and therefore, the staff concludes that these changes are acceptable. Furthermore, the staff finds that these changes meet the intent of NUREG-1431, "Standard Technical Specifications - Westinghouse Plants."

The licensee also proposed to delete footnotes from TS Table 3.3-5 and TS 4.7.1.5 that are no longer applicable. (There is a typographical error in Attachment 1, Page 1 of 5, of the October 1, 1996, letter which incorrectly states that this change is proposed for TS 4.7.6.1.5; rather, it should be TS 4.7.1.5.) The staff considers these changes administrative in nature, and finds them acceptable.

The licensee also proposed changes to TS Table 3.3-4 for the Trip Setpoint and Allowable Value for Item 1.f, "Steam Flow in Two Steam Lines-- High Coincident with Tavg-- Low-Low or Steam Line Pressure-- Low" under the heading "SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION". By letter dated September 4, 1990, the licensee submitted proposed changes to TS Table 3.3-4 to revise the same Trip Setpoint and Allowable Value and it was incorporated into the TSs by Amendment 121 for Unit 1 and Amendment 101 for Unit 2, issued March 11, 1991. However, the changes in those amendments that were made to "Steam Flow in Two Steam Lines-- High Coincident with Tavg-- Low-Low or Steam Line Pressure-- Low" were only made under the heading "STEAM LINE ISOLATION" which is Item 4.d in Table 3.3-4. Since the same signal inputs both "SAFETY INJECTION, TURBINE TRIP AND FEEDWATER" and "STEAM LINE ISOLATION", the September 4, 1990, letter should have proposed changes to both Items 1.f and 4.d of TS Table 3.3-4. Thus, the changes proposed in the October 1, 1996, letter are necessary to make Item 1.f of TS Table 3.3-4 consistent with the plant configuration which had been revised with the implementation of Amendments 121 and 101. This staff finds these changes acceptable since they are needed to make the TSs consistent with the actual plant configuration.

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 and change the surveillance requirements. 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 (61 FR 55040). 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.

Principal Contributor: L. Olshan

Date: January 17, 1997