Docket Nos. 50-272/311 January 2, 1992

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric & Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: INCREASE IN FEEDWATER CONTROL VALVE ISOLATION TIMES, SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M76469 AND M76740)

The Commission has issued the enclosed Amendment Nos. 132 and 111 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 2, 1990.

These amendments increase the allowable isolation times associated with the feedwater control valves and establish consistent isolation times for Salem, Units 1 and 2.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. You are requested to notify the NRC, in writing, when these amendments have been implemented at Salem, Units 1 and 2.

Sincerely,

/S/ Charles L. Miller for

James C. Stone, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 132 to
   License No. DPR-70
2. Amendment No. 111 to
   License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION w/enclosures:

Docket File	MO'Brien(2)	Wanda Jones, 7103	RGoel	
NRC & Local PDR	JStone	CGrimes, 11E-21	KDesai	JWhite, R-I
PDI-2 Reading	JRaleigh	CMcCracken	ACRS(10)	
SVarga	OGC	SNewberry	GPA/PA	
JCalvo	DHagan, MS-3206	RJones	OC/LFMB	
CMiller	GHill(8), P1-37	MWaterman	RBlough, R-I	

OFC	:PDI-2/LA	:PDI-2/PM	:OGC	:PDI-2/D	:
NAME	:MO'Brien	:JStone:rb	:	:CMiller	:
DATE	:12/10/91	:12/10/91	:12/20/91	:1/2/92	:

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

January 2, 1992

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Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
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Public Service Electric & Gas Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: INCREASE IN FEEDWATER CONTROL VALVE ISOLATION TIMES, SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M76469 AND M76740)

The Commission has issued the enclosed Amendment Nos.132 and 111 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 2, 1990.

These amendments increase the allowable isolation times associated with the feedwater control valves and establish consistent isolation times for Salem, Units 1 and 2.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. You are requested to notify the NRC, in writing, when these amendments have been implemented at Salem, Units 1 and 2.

Sincerely,

Charles L. Miller for

James C. Stone, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
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   License No. DPR-70
2. Amendment No.111 to
   License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Steven E. Miltenberger
Public Service Electric & Gas Company

Salem Nuclear Generating Station

cc:

Mark J. Wetterhahn, Esquire
Winston & Strawn
1400 L Street NW
Washington, DC 20005-3502

Richard Fryling, Jr., Esquire
Law Department - Tower 5E
80 Park Place
Newark, NJ 07101

Mr. Calvin A. Vondra
General Manager - Salem Operations
Salem Generating Station
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. S. LaBruna
Vice President - Nuclear Operations
Nuclear Department
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Mr. Thomas P. Johnson, Senior Resident Inspector
Salem Generating Station
U.S. Nuclear Regulatory Commission
Drawer I
Hancocks Bridge, NJ 08038

Dr. Jill Lipoti, Asst. Director
Radiation Protection Programs
NJ Department of Environmental Protection
CN 415
Trenton, NJ 08625-0415

Maryland People's Counsel
American Building, 9th Floor
231 East Baltimore Street
Baltimore, Maryland 21202

Mr. J. T. Robb, Director
Joint Owners Affairs
Philadelphia Electric Company
955 Chesterbrook Blvd., 51A-13
Wayne, PA 19087

Richard B. McGlynn, Commission
Department of Public Utilities
State of New Jersey
101 Commerce Street
Newark, NJ 07102

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Lower Alloways Creek Township
c/o Mary O. Henderson, Clerk
Municipal Building, P.O. Box 157
Hancocks Bridge, NJ 08038

Mr. Frank X. Thomson, Jr., Manager
Licensing and Regulation
Nuclear Department
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. David Wersan
Assistant Consumer Advocate
Office of Consumer Advocate
1425 Strawberry Square
Harrisburg, PA 17120

Mr. Scott B. Ungerer
MGR. - Joint Generation Projects
Atlantic Electric Company
P.O. Box 1500
1199 Black Horse Pike
Pleasantville, NJ 08232

Carl D. Schaefer
External Operations - Nuclear
Delmarva Power & Light Company
P.O. Box 231
Wilmington, DE 19899

Public Service Commission of Maryland
Engineering Division
ATTN: Chief Engineer
231 E. Baltimore Street
Baltimore, MD 21202-3486

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

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. 132
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 April 2, 1990 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. 132 , 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 and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, 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 2, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 132

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 6-15	3/4 6-15

<u>TABLE 3.3-5</u>

<u>ENGINEERED SAFETY FEATURES RESPONSE ITEMS</u>

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
1.	Manual	
a.	Safety Injection (ECCS)	Not Applicable
	Feedwater Isolation	Not Applicable
	Reactor Trip (SI)	Not Applicable
	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
	Auxiliary Feedwater Pumps	Not Applicable
	Service Water System	Not Applicable
	Containment Fan Cooler	Not Applicable
b.	Containment Spray	Not Applicable
	Containment Isolation-Phase "B"	Not Applicable
	Containment Ventilation Isolation	Not applicable
c.	Containment Isolation-Phase "A"	Not Applicable
	Containment Ventilation Isolation	Not Applicable
d.	Steam Line Isolation	Not Applicable
2.	Containment Pressure-High	
a.	Safety Injection (ECCS)	≤27.0(1)
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)

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)	$\leq 27.0^{(1)}/12.0^{(2)}$
b.	Reactor Trip (from SI)	$\leq$ 2.0
c.	Feedwater Isolation	$\leq$ 10.0
d.	Containment Isolation - Phase "A"	$\leq 18.0^{(2)}$
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$ 60
g.	Service Water System	$\leq 49.0^{(1)}/13.0^{(2)}$
4.	Differential Pressure Between Steam Lines-High	
a.	Safety Injection (ECCS)	$\leq 12.0^{(2)}/22.0^{(3)}$
b.	Reactor Trip (from SI)	$\leq$ 2.0
c.	Feedwater Isolation	$\leq$ 10.0
d.	Containment Isolation - Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$ 60
g.	Service Water System	$\leq 13.0^{(2)}/48.0^{(3)}$
5.	Steam Flow in Two Steam Lines - High Coincident with Tavg -- Low-Low	
a.	Safety Injection (ECCS)	$\leq 15.75^{(2)}/25.75^{(3)}$
b.	Reactor Trip (from SI)	$\leq$ 5.75
c.	Feedwater Isolation	$\leq$ 15.0
d.	Containment Isolation - Phase "A"	$\leq 20.75^{(2)}/30.75^{(3)}$
e.	Containment Ventilation Isolation	Not Applicable
f.	Auxiliary Feedwater Pumps	$\leq$ 61.75
g.	Service Water System	$\leq 15.75^{(2)}/50.75^{(3)}$
h.	Steam Line Isolation	$\leq 10.75^{*}$

* $\leq$13.75 until restart following the tenth refueling outage.

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)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 10.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Service Water System	$\leq 14.0^{(2)}/48.0^{(3)}$
h. Steam Line Isolation	$\leq$ 8.0*
7. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	$\leq$ 7.0*
d. Containment Fan Cooler	$\leq$ 40.0
8. Steam Generator Water Level--High High	
a. Turbine Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 10.0
9. Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps(4)	$\leq$ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps(5)	$\leq$ 60.0

* $\leq$10.0 seconds until restart following the tenth refueling outage.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME (Seconds)
D. FEEDWATER ISOLATION		
1. 11 BF 19#	Main Feedwater Isolation	≤9 Sec.
2. 12 BF 19#	Main Feedwater Isolation	≤9 Sec.
3. 13 BF 19#	Main Feedwater Isolation	≤9 Sec.
4. 14 BF 19#	Main Feedwater Isolation	≤9 Sec.
5. 11 BF 40#	Main Feedwater Isolation	≤9 Sec.
6. 12 BF 40#	Main Feedwater Isolation	≤9 Sec.
7. 13 BF 40#	Main Feedwater Isolation	≤9 Sec.
8. 14 BF 40#	Main Feedwater Isolation	≤9 Sec.
E. STEAM GENERATOR BLOWDOWN ISOLATION		
1. 11 GB 4#	Steam Generator Blowdown	≤10 Sec.
2. 12 GB 4#	Steam Generator Blowdown	≤10 Sec.
3. 13 GB 4#	Steam Generator Blowdown	≤10 Sec.
4. 14 GB 4#	Steam Generator Blowdown	≤10 Sec.
5. 11 SS 94#	SG Blowdown Sampling	≤10 Sec.
6. 12 SS 94#	SG Blowdown Sampling	≤10 Sec.
7. 13 SS 94#	SG Blowdown Sampling	≤10 Sec.
8. 14 SS 94#	SG Blowdown Sampling	≤10 Sec.
F. CONTAINMENT PURGE AND PRESSURE - VACUUM RELIEF		
1. 1 VC 1	Purge Supply	≤2 Sec.
2. 1 VC 2	Purge Supply	≤2 Sec.
3. 1 VC 3	Purge Exhaust	≤2 Sec.
4. 1 VC 4	Purge Exhaust	≤2 Sec.
5. 1 VC 5*	Pressure - Vacuum Relief	≤2 Sec.
6. 1 VC 6*	Pressure - Vacuum Relief	≤2 Sec.

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

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. 111
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 April 2, 1990 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. 111 , 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 and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, 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 2, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 111

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 6-17	3/4 6-17

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Service Water System	Not Applicable
Containment Fan Cooler	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
Containment Ventilation Isolation	Not applicable
c. Containment Isolation-Phase "A"	Not Applicable
Containment Ventilation Isolation	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 27.0(1)
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 10.0
d. Containment Isolation-Phase "A"	$\leq$ 17.0(2)/27.0(3)
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	$\leq$ 60
g. Service Water System	$\leq$ 13.0(2)/48.0(3)

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

* $\leq$13.75 seconds until restart following the sixth refueling outage.

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)	$\leq 12.0^{(2)}/22.0^{(3)}$
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 10.0
d. Containment Isolation-Phase "A"	$\leq 17.0^{(2)}/27.0^{(3)}$
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	≤ 60
g. Service Water System	$\leq 14.0^{(2)}/48.0^{(3)}$
h. Steam Line Isolation	≤ 8.0*
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0*
d. Containment Fan Cooler	≤ 40.0
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

* ≤10.0 seconds until restart following the sixth refueling outage.

TABLE 3.6-1 (Contd.)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME (Seconds)
D. FEEDWATER ISOLATION		
1. 21 BF 19#	Main Feedwater Isolation	≤9 Sec.
2. 22 BF 19#	Main Feedwater Isolation	≤9 Sec.
3. 23 BF 19#	Main Feedwater Isolation	≤9 Sec.
4. 24 BF 19#	Main Feedwater Isolation	≤9 Sec.
5. 21 BF 40#	Main Feedwater Isolation	≤9 Sec.
6. 22 BF 40#	Main Feedwater Isolation	≤9 Sec.
7. 23 BF 40#	Main Feedwater Isolation	≤9 Sec.
8. 24 BF 40#	Main Feedwater Isolation	≤9 Sec.
E. STEAM GENERATOR BLOWDOWN ISOLATION		
1. 21 GB 4#	Steam Generator Blowdown	≤10 Sec.
2. 22 GB 4#	Steam Generator Blowdown	≤10 Sec.
3. 23 GB 4#	Steam Generator Blowdown	≤10 Sec.
4. 24 GB 4#	Steam Generator Blowdown	≤10 Sec.
5. 21 SS 94#	SG Blowdown Sampling	≤10 Sec.
6. 22 SS 94#	SG Blowdown Sampling	≤10 Sec.
7. 23 SS 94#	SG Blowdown Sampling	≤10 Sec.
8. 24 SS 94#	SG Blowdown Sampling	≤10 Sec.
F. CONTAINMENT PURGE AND PRESSURE - VACUUM RELIEF		
1. 2 VC 1	Purge Supply	≤2 Sec.
2. 2 VC 2	Purge Supply	≤2 Sec.
3. 2 VC 3	Purge Exhaust	≤2 Sec.
4. 2 VC 4	Purge Exhaust	≤2 Sec.
5. 2 VC 5*	Pressure - Vacuum Relief	≤2 Sec.
6. 2 VC 6*	Pressure - Vacuum Relief	≤2 Sec.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 132 AND 111 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 April 2, 1990, the Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) submitted a request for changes to the Salem Nuclear Generating Station, Unit Nos. 1 and 2, Technical Specifications (TS). The requested changes would increase the allowable isolation times associated with the feedwater control valves in TS Table 3.3-5 and 3.6-1. The changes are proposed due to difficulty in meeting the current TS response time requirements and to be consistent between Units 1 and 2 for functionally identical feedwater systems. Specifically, the licensees propose to increase the response time in Table 3.3-5 from 7 seconds or less to 10 seconds or less for all feedwater isolation functions except for steam flow in two steam lines high coincident with loop average temperature (Tavg) low-low. For the above steam flow in two steam lines, a response time of 15 seconds or less is proposed from the present 10.75 seconds or less because of Tavg total sensor lag time of 5 seconds. In Table 3.6-1, the licensees propose to change the feedwater control valve response time associated with the containment isolation function to 9 seconds or less from the current 5 seconds or less for Unit 1 and 8 seconds or less for Unit 2. The proposed revised closure times acknowledge the time requirements associated with the electronics and ensures that the Engineered Safety Features Actuation System (ESFAS) response time is not exceeded.

2.0 EVALUATION

A. Instrument Response Time

Except for the new Resistance Temperature Detectors (RTDs) installed in the primary coolant system hot legs and cold legs to determine Tavg, the instrument response times are the same as those assumed in the licensees' approved licensing analysis. The licensees state that the Tavg RTDs have a 5 second total sensor lag (response) time. This is consistent with the licensees' submittal to the NRC dated April, 1987 (Ref. 1) which provided supporting documentation for the Salem Unit 1 and 2 RTD Bypass Manifold removal project. As part of its review, the staff found the RTD response time to be acceptable (Ref. 2). Consequently, the staff accepts the licensees' use of 5 seconds for the Tavg RTD response time.

The licensees assume the electronics components have a response time of one second. This assumption is consistent with the value used in the licensees' approved licensing analyses.

The instrument response time and electronics response time portions of the licensees' request for TS revision are consistent with previously approved licensing analyses. Consequently, the staff finds the instrumentation and control systems aspects of the licensees' submittal to be acceptable.

B. Loss of Coolant Accident (LOCA) and Non-LOCA Analysis

Westinghouse performed a safety analysis to determine if an increase in feedwater control valve closure time could be supported by the current licensing basis safety analysis. Westinghouse evaluated the effect of the increase in feedwater valve closure times for LOCA and non-LOCA analyses. In addition, an analysis of the consequences of a complete failure of a feedwater control valve to close was also performed by Westinghouse.

(1) Increase in feedwater valve closure time.

During small and large break LOCAs, an extension in the time required to isolate feedwater would increase the decay heat removal capability slightly and result in a small benefit during these events. The failure of a feedwater control valve to close results in the same small benefit and it is bounded by the single failure assumed in the Salem licensing basis LOCA analysis.

Past analyses performed for steamline break core protection purposes indicate that a small increase in core power (maximum of 1%) would result due to the increase in feedwater control valve closure time. The departure from nucleate boiling ratio (DNBR) penalty associated with this slight core power increase does not exceed the design limit value of DNBR. Thus, the consequences and conclusions of the existing Salem steamline break core protection analysis are still applicable.

(2) Failure of a feedwater control valve to close. (This is an additional evaluation performed by Westinghouse for this amendment.)

The design basis steamline break core analysis currently assumes the limiting single failure of a safeguard train, which minimizes the boron injection capability to terminate the event. If the single failure was assumed to be the failure of a feedwater control valve to close, a 30-second delay in feedwater isolation would be imposed because this is the closure time for the feedwater isolation valve which is in series with the feedwater control valve. Continued feedwater addition at a rate of 125% of full feedwater flow for 30 seconds was evaluated. The results showed that the positive reactivity insertion resulting from the additional cooldown prior to feedwater isolation would be less than the negative reactivity from boron injection provided by a second safeguard train. Therefore, the single failure of the feedwater control valves to close would be less limiting than the failure of a safeguard train.

In summary, the conclusions of the current Salem licensing basis analyses for LOCA and non-LOCA events would be unchanged if the feedwater isolation ESFAS response time was increased as proposed. The single failure of a feedwater control valve to close is bounded by the single failure assumptions used for the Salem licensing basis LOCA and non-LOCA related analyses.

Based on the licensees' evaluation of LOCA and non-LOCA events for an increase in feedwater isolation control valve isolation response times, the staff concludes that the proposed TS changes are acceptable.

C. <u>Containment Integrity Analysis</u>

The licensees indicated that the Salem design basis containment analyses considered the short and long-term mass and energy release for postulated LOCAs, containment response analyses following a LOCA or steamline break inside containment, and subcompartment pressure transient analyses.

The licensees stated that increasing the feedwater control valve closure time would have no effect on the calculated results for short-term mass and energy release and subcompartment pressure analyses because the transient has a duration of 3 seconds or less. The long-term mass and energy release and containment pressure response following a LOCA would improve with increased feedwater isolation closure times because of the reduction in steam generator secondary side temperature as the mass increases and thus it will reduce secondary to primary heat transfer occurring during a LOCA. The staff agrees with the above discussion that the increased closure time will have no negative effect on the short-term and long-term mass and energy releases, short-term subcompartment analysis and the containment pressure response following a postulated LOCA.

The licensees indicated that the increase in valve closure time can affect the steamline break containment analysis slightly. The current Salem design basis containment analysis include multiple failure assumptions. The existing most limiting containment pressure occurs for a 0.944 square feet split rupture at 30% power, with the failure of a main steam isolation valve (MSIV) and a containment safeguard train, resulting in a peak pressure of 46.4 psig. The most limiting analyses were reevaluated with the feedwater closure time increased to 10 seconds with all single failures. This resulted in a peak pressure of 46.53 psig. Therefore, the containment pressure will be maintained below the design pressure of 47 psig for all single failures analyzed.

The licensees also indicated that the existing most limiting containment temperature occurs for a 0.6 square feet double ended rupture initiated at 102% power, with failures of an MSIV, feedwater control valve, feedwater pump runout protection, and a containment safeguard train. The associated peak temperature is 345.5 °F. The most limiting analyses were reevaluated with feedwater control valve closure time increased to 10 seconds with all single failures. This resulted in a peak temperature of 338.3 °F. Therefore, the containment temperature will be maintained below 340 °F for all single failures analyzed.

Based on the above discussion, the staff agrees that the proposed increase in feedwater control valve closure time does not affect the containment integrity as the containment design pressure and temperature will not be exceeded.

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 (56 FR 51930). 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 Contributors:
M. Waterman, SICB
R. Goel, SPLB
K. Desai, SRXB

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REFERENCES

1. Licensing Report S-87-05, "Licensing Report for New Narrow Range Temperature Measurement System (RTD Bypass Elimination, PSE&G, Salem, Units 1 and 2," April, 1987.

2. Safety Evaluation that accompanied Amendments 84 and 56 for Salem, Units 1 and 2, respectively, "Technical Specification Changes Due to RTD Bypass System Modifications" dated November 16, 1987.