June 4, 19

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. M97827 AND M97828)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 194 and 177 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 January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997.

These amendments provide changes to Technical Specification (TS) 3.4.3, "Relief Valves," for Salem Unit 1, and TS 3.4.5, "Relief Valves," for Salem Unit 2, to ensure that the automatic capability of the power operated relief valves to relieve pressure is maintained when these valves are isolated by closure of the block valves.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be published in the Federal Register.

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. 194 to License No. DPR-70
2. Amendment No. 177 to License No. DPR-75
3. Safety Evaluation
4. Notice of Issuance

cc w/encls: See next page

DFol

DISTRIBUTION

Docket File	JStolz	GHill(4)	JLinville, RGN-I	GWest
PUBLIC	MO'Brien(2)	JLyons	DJaffe	
PDI-2 Reading	LOlshan	CGrimes	CLiang	
SVarga	OGC	ACRS	SMazumdar	

OFFICE	PDI-2/PM	PDI-2/LA	SRXB	OGC	PDI-2/D
NAME	LOlshan:rb	MO'Brien	JLyons	with changes	JStolz
DATE	5/16/97	5/16/97	5/19/97	5/28/97	6/04/97

OFFICIAL RECORD COPY
DOCUMENT NAME: SA97827.AMD

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

June 4, 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. M97827 AND M97828)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 194 and 177 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 January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997.

These amendments provide changes to Technical Specification (TS) 3.4.3, "Relief Valves," for Salem Unit 1, and TS 3.4.5, "Relief Valves," for Salem Unit 2, to ensure that the automatic capability of the power operated relief valves to relieve pressure is maintained when these valves are isolated by closure of the block valves.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be published in the Federal Register.

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. 194 to License No. DPR-70
2. Amendment No. 177 to License No. DPR-75
3. Safety Evaluation
4. Notice of Issuance

cc w/encls: See next page

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

Salem Nuclear Generating Station, Units 1 and 2

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. D. F. Garchow
General Manager - Salem Operations
Salem Generating Station
P.O. Box 236
Hancocks Bridge, NJ 08038

Mr. Louis Storz
Sr. Vice President - Nuclear Operations
Nuclear Department
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Mr. Charles S. Marschall, Senior Resident Inspector
Salem Generating Station
U.S. Nuclear Regulatory Commission
Drawer 0509
Hancocks Bridge, NJ 08038

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

Maryland Office of People's Counsel
6 St. Paul Street, 21st Floor
Suite 2102
Baltimore, Maryland 21202

Ms. R. A. Kankus
Joint Owner Affairs
PECO Energy Company
965 Chesterbrook Blvd., 63C-5
Wayne, PA 19087

Mr. Elbert Simpson
Senior Vice President - Nuclear Engineering
Nuclear Department
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Richard Hartung
Electric Service Evaluation
Board of Regulatory Commissioners
2 Gateway Center, Tenth Floor
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. David R. Powell, Manager
Licensing and Regulation
Nuclear Business Unit
P.O. Box 236
Hancocks Bridge, NJ 08038

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

P. M. Goetz
MGR. Joint Generation
Atlantic Energy
6801 Black Horse Pike
Egg Harbor Twp., NJ 08234-4130

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

Public Service Commission of Maryland
Engineering Division
Chief Engineer
6 St. Paul Centre
Baltimore, MD 21202-6806

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. 194
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 January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, 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. 194 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended by changes to Appendix C as indicated in the attachment to this license amendment, and paragraph 2.C.(10) of the Facility Operating License No. DPR-70 is amended to read as follows:

(10) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 194 , are hereby incorporated into this license. Public Service Electric and Gas Company shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of its date of issuance and the change to the facility shall be implemented prior to entry into Mode 3 from the current outage for Salem Unit 1. Implementation of this amendment shall include upgrading the initiation circuitry for the power operated relief valves as described in the licensee's application dated January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, and evaluated in the staff's safety evaluation attached to this amendment.

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

Attachments: 1. Page 1 to Appendix C of License* DPR-70
2. Changes to the Technical Specifications

Date of Issuance: June 4, 1997

* Page 1 of Appendix C is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 194

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

1. Remove — Insert

Remove	Insert
Appendix C, page 1	Appendix C, page 1

2. Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-5	3/4 4-5
B 3/4 4-1a	B 3/4 4-1a
-	B 3/4 4-1b

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. DPR-70

Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company, and Atlantic City Electric Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
192	The licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated January 11, 1996, as supplemented by letters dated February 26, May 22, June 27, July 12, December 23, 1996, and March 17, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from March 21, 1997.
194	The licensee is authorized to upgrade the initiation circuitry for the power operated relief valves, as described in the licensee's application dated January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented prior to entry into Mode 3 from the current outage for Salem, Unit 1.

3/4.4.3 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3 Two power relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 6 hours either restore at least one PORV to OPERABLE status or close the associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Restore the remaining PORV to OPERABLE status within 72 hours from failure of that valve to meet the Limiting Condition for Operation.

d. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place the associated PORV in manual control; restore the block valve to operable status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With both block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place the associated PORVs in manual control; restore at least one block valve to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Restore the remaining block valve to OPERABLE status within 72 hours from failure of that valve to meet the Limiting Condition for Operation.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 pounds per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperature. While in Mode 5 the safety valve requirement may be met by establishing a vent path of equivalent relieving capacity when no code safety valves are OPERABLE.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 RELIEF VALVES

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

B. Automatic control of PORVs to control reactor coolant system pressure. This is a function that reduces challenges to the code safety valves for overpressurization events, including an inadvertent actuation of the Safety Injection System.

C. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

3/4.4.3 RELIEF VALVES (continued)

D. Manual control of the block valve to : (1) unblock an isolated PORV to allow it to be used for manual and automatic control of Reactor Coolant System pressure (Items A & B), and (2) isolate a PORV with excessive seat leakage (Item C).

E. Manual control of a block valve to isolate a stuck-open PORV.

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. 177
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 January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, 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. 177 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended by changes to Appendix C as indicated in the attachment to this license amendment, and paragraph 2.C.(26) to Facility Operating License No. DPR-75 is amended to read as follows:

(26) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 177 , are hereby incorporated into this license. Public Service Electric and Gas Company shall operate the facility in accordance with the Additional Conditions.

3. This amendment is effective as of its date of issuance and the change to the facility shall be implemented prior to entry into Mode 3 from the current outage for Salem Unit 2. Implementation of this amendment shall include upgrading the initiation circuitry for the power operated relief valves as described in the licensee's application dated January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, and evaluated in the staff's safety evaluation attached to this amendment.

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

Attachments: 1. Page 1 to Appendix C of License* DPR-75
2. Changes to the Technical Specifications

Date of Issuance: June 4, 1997

* Page 1 of Appendix C is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 177

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

1. Remove | Insert

Remove	Insert
Appendix C, page 1	Appendix C, page 1

2. Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-8	3/4 4-8
B 3/4 4-2	B 3/4 4-2
B 3/4 4-3	B 3/4 4-3
-	B 3/4 4-3a

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. DPR-75

Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company, and Atlantic City Electric Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
175	The licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated January 11, 1996, as supplemented by letters dated February 26, May 22, June 27, July 12, December 23, 1996, and March 17, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented within 60 days from March 21, 1997.
177	The licensee is authorized to upgrade the initiation circuitry for the power operated relief valves, as described in the licensee's application dated January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, and evaluated in the staff's safety evaluation attached to this amendment.	The amendment shall be implemented prior to entry into Mode 3 from the current outage for Salem, Unit 2.

3/4.4.5 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.5 Two power relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 6 hours either restore at least one PORV to OPERABLE status or close the associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Restore the remaining PORV to OPERABLE status within 72 hours from failure of that valve to meet the Limiting Condition for Operation.

d. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place the associated PORV in manual control; restore the block valve to operable status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With both block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place the associated PORVs in manual control; restore at least one block valve to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Restore the remaining block valve to OPERABLE status within 72 hours from failure of that valve to meet the Limiting Condition for Operation.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 pounds per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperature. While in Mode 5 the safety valve requirement may be met by establishing a vent path of equivalent relieving capacity when no code safety valves are OPERABLE.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip setpoint is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control RCS pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown.

3/4.4.5 RELIEF VALVES (continued)

B. Automatic control of PORVs to control reactor coolant system pressure. This is a function that reduces challenges to the code safety valves for overpressurization events, including an inadvertent actuation of the Safety Injection System.

C. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

D. Manual control of the block valve to : (1) unblock an isolated PORV to allow it to be used for manual and automatic control of Reactor Coolant System pressure (Items A & B), and (2) isolate a PORV with excessive seat leakage (Item C).

E. Manual control of a block valve to isolate a stuck-open PORV.

3/4.4.6 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

3/4.4.6 STEAM GENERATORS (continued)

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 194 AND 177 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 January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, 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 change Technical Specification (TS) 3.4.3, "Relief Valves," for Salem Unit 1, and TS 3.4.5, "Relief Valves," for Salem Unit 2, to ensure that the automatic capability of the power operated relief valves (PORVs) to relieve pressure is maintained when these valves are isolated by closure of the block valves. The letters dated April 8 and 28, 1997, contained supplementary and clarifying information that did not expand the scope of the April 4, 1997 (62 FR 16199), Federal Register notice.

2.0 DESCRIPTION

In response to the Westinghouse Nuclear Safety Advisory letter (NSAL), NSAL 93-013, the licensee has determined that an inadvertent Safety Injection (SI) actuation at power could cause the pressurizer to become water solid and pressurizer safety valves lifting with water relief if the automatic operation of the PORVs is not made available for reactor coolant system depressurization early in the transient. The Salem pressurizer safety valves are not designed to relieve water. Thus, the water relief has the potential to cause the pressurizer safety valves to fail in the open position.

In the course of the review of the licensee's January 31, 1997 application, the NRC Staff noted that the pressurizer PORVs were not designed to "safety related" standards and thus could not be credited for mitigation of the inadvertent SI actuation at power incident when the PORV is operating in the automatic mode. In response to this observation, the licensee proposed an upgrade of PORVs as described in the March 14, 1997 and April 8, 1997 supplements, to eliminate the possibility that a single active failure of a PORV component could prevent the mitigation of the inadvertent SI actuation at power incident. In addition, because the licensee relied upon reactor operator action to assure that the PORV Block Valves are open in the event of an SI actuation at power incident, the NRC staff requested additional information regarding reactor operator performance. The licensee provided additional information regarding reactor operator performance in the licensee's letter dated April 8, 1997.

3.0 EVALUATION

The existing electrical and control system associated with the automatic operation of the PORVs is designed to control grade standard without protection from single failures. In order to take credit for the PORVs' automatic function for mitigating the inadvertent SI actuation event, the licensee in letters dated March 14, 1997 and April 8, 1997 proposed modification to the PORV circuitry to eliminate single failure vulnerabilities in the PORV circuitry and upgrade circuitry to qualify the PORVs as safety-related.

3.1 Upgrade of PORV Circuitry

A PSE&G analysis determined that, although in the event of an inadvertent SI at power, credit could be taken for unblocking one PORV to mitigate the consequences of the event, it could not be ascertained that only manual cycling of the PORV(s) would ensure maintaining reactor coolant system (RCS) pressure below the pressurizer safety relief valve setpoint in all cases. Furthermore, the PSE&G analysis demonstrated better reliability from automatic operation of the PORVs to mitigate the consequences of an inadvertent SI at power event. Therefore, PSE&G proposed a number of modifications to the PORV circuitry to eliminate single failure vulnerabilities and upgrade the PORV control system to qualify as safety-related.

The proposed modifications to the PORV circuitry are intended to: 1) eliminate the non-safety-related controller, PC455K, from the valve control circuitry and relocate the PORV circuitry into the protection racks from the control racks; 2) separate the PORV control channels such that failure of any one channel or power supply will render, at most, one PORV inoperable; 3) replace each high pressure Hagan comparator (2PC455A, 2PC456A, 2PC457A, and 2PC474C) with a dual Hagan comparator; 4) remove comparators 2PC455E, 2PC457E, and 2PC474B and replace comparators 2PC456E and F with a single channel comparator since functions of these comparators will be performed by the dual comparators described in 3); 5) add four overhead annunciators to indicate a PORV unsafe condition; and 6) add new bistable test switches.

The NRC staff concludes that the above modifications to the PORV circuitry will eliminate single failure vulnerabilities, qualify the PORVs in accordance with IEEE Standard 279, and that the upgraded circuitry qualifies the PORVs as safety-related.

3.2 PORV Performance

PSE&G evaluated the capacity of the PORV air accumulators and concluded that the current accumulators are sufficient to operate the PORVs for about 45 minutes which is sufficient time for the operators to manually terminate the inadvertent SI actuation at power transient. The licensee has stated that the inadvertently actuated SI flow could be terminated within 25 minutes into the transient using plant Emergency Operating Procedures (EOPs) and during this event the PORVs will experience approximately 220 full strokes. In the January 31, 1997 application, the licensee indicated that each PORV with two dedicated air accumulators can perform 305 full stroke open and close operations in the event that the normal supply of air is unavailable. After 305 full strokes, the PORVs will be able to perform an additional 486 50% strokes. In the April 28, 1997 supplement, the licensee indicated that a reanalysis of the PORV air supply indicated that approximately 350 PORV strokes would be available (145 full strokes and 205 partial [50%] strokes). The revised PORV performance assessment is acceptable in that more than 220 valve strokes would be available assuming that the normal air source is unavailable. The licensee further indicated that the relieving capacity of a 50% stroke is essentially equivalent to a full stroke and is considered as equivalent for the purpose of determining the adequacy of PORV availability. The licensee has concluded that the PORVs will function adequately under these operating conditions.

Endurance tests performed with five different trims (with different trim materials) on one PORV at Wyle Laboratories demonstrated that: 1) after 2000 consecutive operations, there were no packing leaks nor packing gland adjustments required; 2) there was no diaphragm failure; and 3) the solenoid valve withstood 10,000 operations without any loss of function.

The staff concludes that the PORV performance is acceptable with regard to its expected performance in the mitigation of the inadvertent SI actuation at power event.

3.2 Reactor Operator Performance

The licensee has re-analyzed the inadvertent SI at power event based on the use of EOPs to make PORVs available by opening their associated block valve within 10 minutes into the transient. This assumption has been validated by simulator test results which indicate that the operators have been successful in accomplishing this procedure within seven to nine minutes. The Salem operators are trained for these EOPs.

With regard to operator performance, in those instances where licensees consider temporary or permanent changes to the facility which eliminate operator actions, where prior credit for operator actions was taken, the staff has relied on the guidance provided in Generic Letter (GL) 91-18, and ANSI/ANS 58.8, "Time Response Design Criteria for Safety Related Operator Actions," 1984 (ANSI-58.8), for evaluating such changes. While ANSI-58.8 supplies estimates of reasonable response times for operator actions, the standard does allow licensees to use time intervals derived from independent sources.

The NRC Staff reviewed specific operator actions and the times required for these actions. The licensee stated the operator is expected to act within about 10 minutes to open at least one pressurizer PORV by opening its associated block valve, which precludes the potential for steam or water relief through the pressurizer safety valve. The licensee noted that to perform this action, the operator depresses a bezel button. In addition, the licensee stated that the range of times for operating crews to perform the required manual action was seven to nine minutes.

With regard to procedural guidance for required actions, the licensee submitted plant procedure EOP-TRIP-1, step 23, which documented the required local manual actions. The licensee pointed out that after the PORV block valve is opened, the bezel indication light will illuminate, thus indicating that this step has been successfully completed.

The licensee stated that operator training to carry out the required actions included knowledge-based training (control room manipulation is a license-basis critical time) and skill-based training (PORV manipulation is performed on the Salem site-specific simulator).

The NRC staff reviewed the ability to recover from plausible errors in performance of manual actions associated with PORV block valve manipulations, and the expected time required to make such a recovery. The licensee's evaluation did not consider the possibility of performance errors or the likelihood of recovering from such errors given the timeframe (i.e., 10 minutes) allotted to accomplish the manual action; however, given the time assumed for operator action, it is likely that recovery from an error in performance would be achieved.

The staff finds the previously discussed information acceptable in that it is consistent with ANSI-58.8 and Generic Letter 91-18.

3.3 Technical Specifications

The licensee proposed TSs regarding PORVs would assure the operability of the PORVs for their automatic and manual operating function. Also, the proposed TSs are consistent with the recommendation of Generic Letter 90-06, "Resolution of Generic Issue 70, 'PORV and Block Valve Reliability,' and 94, 'Additional LTOP [Low Temperature Overpressure Protection] Protection for PWRs' [Pressurized Water Reactors]," dated June 25, 1990, and therefore acceptable. The proposed TS bases provide clarifications of the safety-related function to be performed by the PORVs including the use of the automatic function of the PORVs to mitigate an inadvertent actuation of SI event. We have reviewed the licensee's submittal and find the proposed changes acceptable.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. In an April 15, 1997, telephone conversation with the State official, Mr. R. Pinney, noted that the licensee's proposed TS Bases for the PORV TSs did not conform to the format suggested by the NRC staff's Standard Technical Specifications (STS).

The requirements regarding the content of the TSs are contained in 10 CFR 50.36 (Title 10 of the Code of Federal Regulations, Part 50, Section 36), "Technical specifications." In Subsection (a) of 10 CFR 50.36, licensees are required to provide, "A summary statement of the bases or reasons for such specifications...but [they], shall not become part of the technical specifications." While the NRC staff requires the TS Bases to be technically accurate, the licensee is free to otherwise change the Bases without prior NRC approval.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on May 9, 1997 (62 FR 25675). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 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: D. Jaffe
C. Liang
S. Mazumdar
G. West

Date: June 4, 1997

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos.194 and 177 to Facility Operating License Nos. DPR-70 and DPR-75, respectively, to Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees), which revised the Technical Specifications for operation of the Salem Nuclear Generating Station, Units 1 and 2, located at the licensee's site in Salem County, New Jersey.

The amendments provide changes to Technical Specification (TS) 3.4.3, "Relief Valves," for Salem Unit 1, and TS 3.4.5, "Relief Valves," for Salem Unit 2, to ensure that the automatic capability of the power operated relief valves to relieve pressure is maintained when these valves are isolated by closure of the block valves.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on April 4, 1997 (62 FR 16199). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (62 FR 25675).

For further details with respect to the action see (1) the application for amendment dated January 31, 1997, as supplemented by letters dated March 14, April 8, and April 28, 1997, (2) Amendment Nos.194and177to License Nos. DPR-70 and DPR-75, respectively, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room located at the Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

Dated at Rockville, Maryland, this 4th day of June 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

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