November 16, 1987

Docket Nos. 50-272/311

Mr. Corbin A. McNeill, Jr.
Senior Vice President - Nuclear
Public Service Electric & Gas Company
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

SUBJECT: TECHNICAL SPECIFICATION CHANGES DUE TO RTD BYPASS SYSTEM MODIFICATIONS (TAC NOS. 64857 AND 64858)

Re: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

The Commission has issued the enclosed Amendment Nos. 84 and 56 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 May 5, 1987 and supplemented by letters dated September 2, 1987 and October 1, 1987, which contained confirmatory and clarifying details.

These amendments change the Technical Specifications due to a modification of the reactor trip system and engineered safety features response times to accommodate the removal of the RTD bypass system.

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/

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

Enclosures:
1. Amendment No. 84 to License No. DPR-70
2. Amendment No. 56 to License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	MO'Brien (2)	Wanda Jones	BClayton
NRC PDR	OGC - Bethesda	EButcher	RGallo
Local PDR	DHagan	Tech Branch	
PDI-2 Reading	EJordan	ACRS (10)	
WButler	JPartlow	CMiles, GPA/PA	
DFischer/MThadani	TBarnhart (8)	RDiggs, ARM/LFMB	

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	DFischer	S H Lewis	WButler
11/11/87	11/13/87	11/5/87	11/16/87

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

November 16, 1987

Docket Nos. 50-272/311

Mr. Corbin A. McNeill, Jr.
Senior Vice President - Nuclear
Public Service Electric & Gas Company
P.O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

SUBJECT: TECHNICAL SPECIFICATION CHANGES DUE TO RTD BYPASS SYSTEM MODIFICATIONS (TAC NOS. 64857 AND 64858)

Re: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

The Commission has issued the enclosed Amendment Nos. 84 and 56 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 May 5, 1987 and supplemented by letters dated September 2, 1987 and October 1, 1987, which contained confirmatory and clarifying details.

These amendments change the Technical Specifications due to a modification of the reactor trip system and engineered safety features response times to accommodate the removal of the RTD bypass system.

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

Sincerely,

Donald C. Fischer

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

Enclosures:
1. Amendment No. 84 to License No. DPR-70
2. Amendment No. 56 to License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. C. A. McNeill
Public Service Electric & Gas Company

Salem Nuclear Generating Station

cc:

S. E. Miltenberger
Vice President - Nuclear Operations
Nuclear Department
P.O. Box 236
Hancocks Bridge, NJ 08038

Mark J. Wetterhahn, Esquire
Conner and Wetterhahn
Suite 1050
1747 Pennsylvania Avenue, NW
Washington, DC 20006

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

Mr. John M. Zupko, Jr.
General Manager - Salem Operations
Salem Generating Station
Post Office Box E
Hancocks Bridge, NJ 08038

Robert Traae, Mayor
Lower Alloways Creek Township
Municipal Hall
Hancocks Bridge, NJ 08038

Thomas Kenny, Resident Inspector
Salem Nuclear Generating Station
U.S. Nuclear Regulatory Commisstion
Drawer I
Hancocks Bridge, NJ 08038

Richard F. Engel
Deuputy Attorney General
Department of Law and Public Safety
CN-112
State House Annex
Trenton, NJ 08625

David M. Scott, Acting Chief
Bureau of Nuclear Engineering
Department of Environmental Protection
CN 411
Trenton, NJ 08625

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
631 Park Avenue
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. Bruce A. Preston, Manager
Licensing and Regulation
Nuclear Department
P.O. Box 236
Hancocks Bridge, NY 08038

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

Morgan J. Morris, III
General Manager - Operating License
Atlantic Electric
P.O. Box 1500
1199 Black Horse Pike
Pleasantville, NJ 08232

Delmarva Power
c/o Thomas S. Shaw, Jr.
Vice President - Production
800 King Street
P.O. Box 231
Wilmington, DE 19899

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. 84
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 May 5, 1987 and supplemented by letters dated September 2, 1987 and October 1, 1987, 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. 84 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Donald C. Fischer for
Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 16, 1987

Previously concurred*

PDI-2/LA	PDI-2/PM*	OGC*	PDI-2/D*
MO'Brien	DFischer	SHLewis	WButler
11/16/87	11/03/87	11/05/87	11/16/87

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 84 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/s /

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 16, 1987

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	DFischer	S H Lewis	WButler
/ /87	11/3/87	11/5/87	11/16/87

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 84 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Donald C. Fischer /for

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 84

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-5	2-5
2-9	2-9
3/4 3-9	3/4 3-9
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 3-30	3/4 3-30
3/4 3-31	3/4 3-31

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 87,300 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T'') - f_2(\Delta I)]$

where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, °F
T''	=	Reference T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F
K_4	=	1.080
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.00119/°F for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.
S	=	Laplace transform operator, Sec^{-1}.
$f_2(\Delta I)$	=	0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.1 percent.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.0 percent.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 5.75 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE ITEMS

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	$\leq$ 7.0
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	$\leq$ 7.0
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	$\leq$ 7.0
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	$\leq$ 10.75
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

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	≤ 7.0
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	≤ 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	≤ 11.0
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 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Undervoltage RCP bus	
a. Turbine-Driven Auxiliary Feedwater Pumps	$\leq$ 60.0
11. Containment Radioactivity - High	
a. Purge and Pressure Vacuum Relief	$\leq$ 5.0(6)
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	$\leq$ 4.0
14. Station Blackout	
a. Motor-Driven Auxiliary Feed Pumps	$\leq$ 60.0

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) On 2/3 in any steam generator.

(5) On 2/3 in 2/4 steam generators.

(6) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time the Isolation Valves are fully shut.

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. 56
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 May 5, 1987 and supplemented by letters dated September 2, 1987 and October 1, 1987, 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. 56 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Donald C. Fischer for
Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1987

Previously concurred*

PDI-2/LA	PDI-2/PM*	OGC*	PDI-2/D*
MO'Brien	DFischer	SHLewis	WButler
11/16/87	11/03/87	11/05/87	11/16/87

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 56 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1987

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	DFischer	SHLewis	WButler
/ /87	11/3/87	11/5/87	11/16/87

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 56 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Donald C. Fischer /for

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 56

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows. Asterisk pages are provided to maintain document completeness.

Remove Pages	Insert Pages
2-5	2-5
2-9	2-9
3/4 3-9	3/4 3-9
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 3-31	3/4 3-31
3/4 3-32	3/4 3-32

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 25% of RATED THERMAL POWER High Setpoint - $\leq$ 109% of RATED THERMAL POWER	Low Setpoint - $\leq$ 26% of RATED THERMAL POWER High Setpoint - $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux, High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 second	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 second
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	$\geq$ 1865 psig	$\geq$ 1855 psig
10. Pressurizer Pressure--High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level--High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Loss of Flow	$\geq$ 90% of design flow per loop*	$\geq$ 89% of design flow per loop*

*Design flow is 87,300 gpm per loop.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o [K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T - T") - f_2(\Delta I)]$

where:

ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

$T"$ = Reference T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

K_4 = 1.080

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature

K_6 = 0.00119/°F for $T > T"$; $K_6 = 0$ for $T \leq T"$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator, Sec^{-1}.

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 3.1 percent.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.0 percent.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	NOT APPLICABLE
7. Overtemperature ΔT	$\leq$ 5.75 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	NOT APPLICABLE

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

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)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
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	$\leq 13.0^{(2)}/48.0^{(3)}$

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	$\leq$ 7.0
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	$\leq$ 7.0
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	$\leq$ 10.75
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

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	≤ 7.0
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	≤ 11.0
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 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Undervoltage RCP bus	
a. Turbine-Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Containment Radioactivity - High	
a. Purge and Pressure Vacuum Relief	≤ 5.0(6)
12. Trip of Feedwater Pumps	
a. Auxiliary Feedwater Pumps	Not Applicable
13. Undervoltage, Vital Bus	
a. Loss of Voltage	≤ 4.0
14. Station Blackout	
a. Motor-Driven Auxiliary Feed Pumps	≤ 60.0

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, SI and RHR pumps.

(2) Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(3) Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

(4) On 2/3 in any steam generator.

(5) On 2/3 in 2/4 steam generators.

(6) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time the Isolation Valves are fully shut.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 84 AND 56 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 May 5, 1987, and supplemented by letters dated September 2, 1987 and October 1, 1987, Public Service Electric & Gas Company (PSE&G), the licensee, requested an amendment to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. The proposed amendments would change the Technical Specifications due to modification of the reactor trip system and engineered safety features response times to accommodate the removal of the RTD bypass system. A new reactor coolant system (RCS) temperature measurement system would be installed in place of the RTD bypass system. This system will use narrow range dual element mounted resistance temperature detectors (RTDs). This design modification is to overcome major drawbacks of the RTD bypass system which lacked reliability (leakage from valve packing or mechanical joints) and resulted in high radiation doses during the performance of maintenance around the RTD bypass system.

The licensee's supplementary submittals of September 2 and October 1, 1987 were made as a result of an NRC staff request to correct and clarify the language of the original submittal, and do not contain substantive changes.

2.0 EVALUATION AND SUMMARY

The new method proposed for measuring the hot and cold leg temperatures uses narrow-range, dual element, fast response RTDs manufactured by the Weed Company. One of each of the RTD dual elements is used while the other is installed as a spare. The RTDs are placed in thermowells to allow replacement without draindown. The thermowells, however, increase the response time.

The three RTDs in each of the hot legs are placed within the three existing scoops. Outlet ports are provided in the scoops to direct the sampled fluid past the sensing element of the RTDs. This method of measuring the hot leg temperature by scoop mixing was conceived by Combustion Engineering (CE) and is a proprietary design. Since there is no temperature streaming problem in the cold leg, only one dual element RTD is installed in a thermowell associated with the cold leg to provide the cold leg temperature reading.

Because of the variation in temperature in the cross-section of the hot legs due to hot leg streaming, the three RTD measurement locations in each hot leg are used to get an average value of the variation. An electronic system is used to perform the averaging of the reactor coolant hot leg signals from the three RTDs in each hot leg and then to transmit the signal for the average hot leg temperature to protection and control systems. There is a routine for performing a quality check of the three temperature signals for each hot leg. A failed RTD would be picked up by the T_{ave} or delta T deviation alarm. Also, each channel is checked every eight hours. On failure of a RTD, the channel would be tripped and the Technical Specifications action statement would go into effect. The second element of each RTD is a spare and its leads can be switched from the failed RTD leads in the control room instrument panel.

The overall response time of the proposed thermowell RTD hot leg temperature system (6.0 seconds) has been designed to remain the same as in the former RTD bypass system (6.0 seconds). The licensee has reported that the combined RTD/thermowell response time of 4.75 seconds for the proposed system is conservative as the RTD instrument specification requires that both elements be less than 4.0 seconds and typical results for the same model Weed RTD in CE plant thermowells have demonstrated that response times less than 4.0 seconds are realistic. The licensee has reported that the response times will be checked as part of the reactor trip system instrumentation (Technical Specification Item 7, Table 3.3-2) and engineered safety features response time (Item 5, Table 3.3-5). The surveillance requirements state that response time checks are required at each refueling. RTD response times have been known to degrade and Loop Current Step Response (LCSR) methodology is the recommended on-site method for checking RTD response times. The licensee plans to use the LCSR method for checking the RTD response time at each refueling.

Based on the above information the staff finds that the RTD response time has been addressed in an acceptable manner.

The new method of measuring each hot leg temperature with three thermowell RTDs, used in place of the RTD bypass system with three scoops, has been analyzed to be slightly more accurate than with the RTDs in the existing bypass system. As previously mentioned, the scoops are used to obtain a sampling of the flow (five holes in each scoop) at three 120 degree sectors in each of the hot legs in order to obtain a more accurate hot leg average temperature that accounts for the non-uniform temperature streaming. Previously, the RTD bypass system took the sampled flows from the scoops and made an external RTD temperature measurement in a plenum section. The new method with the RTD bypass system removed will measure the sampled mixed coolant flow with a dual element Weed RTD mounted in a thermowell. The Weed RTD is mounted in the existing scoop near new outlet ports in the scoop. This proprietary design has been evaluated by the licensee. A model test has been completed and calculations performed to ascertain that an accurate mixed mean temperature will be measured. The model test provided information for the selection of the proper location of the RTD sensor in the scoop for accurate measurement and the expected temperature bias. The licensee has made a commitment to obtain confirmatory information on the mixed mean temperature accuracy. This will be done by comparing pre-installation and post-installation calorimetric data on the RTD temperature measurements in the Salem plant for matching operating conditions. The licensee will make these data available to the staff.

The dual element Weed RTD has improved accuracy over the existing RTDs. The total uncertainty is ± 0.7°F. This value includes a drift (for 22.5

months) of + 0.4°F on top of the normal ± 0.3°F accuracy (includes hysteresis and repeatability). For the hot leg temperature measurement, there is a need to apply a small temperature bias. This temperature bias is based on the model test information which identified a scoop RTD installation location effect for the hot leg temperature measurement.

Because three RTDs are used to measure each hot leg temperature instead of the former single measurement, the error associated with the hot leg measurement is reduced to one over the square root of three compared to a single RTD. The impact of the additional electronics needed for the two additional hot leg RTD's per loop has been found by the licensee to be minimal.

The three RTD signals are averaged to obtain the loop's T_{hot} value. The existing overall channel functional checks and calibration accuracy requirements are to be maintained. The impact of the rack drift has been considered in the evaluation.

There is no change to the cold leg's electronics. Therefore, there is no impact to the cold leg accuracy other than the increase obtained from the more accurate RTD.

The licensee intends to replace 2 RTDs per refueling on the lead unit at each of the following two refueling outages. They will review the recalibration results from the RTDs removed, as well as other data anticipated to become available on the drift of the Weed RTDs, prior to making any subsequent long-range periodic RTD replacements. Since the replacement RTD would have to be within the allowable deviation from the averaged reading, verification of no significant systematic drift will be obtained.

The net result of the proposed RTD bypass system modification is a slight improvement in the accuracy of the temperature related functions over the accuracy now achievable with the existing RTD's in the bypass system. The licensee has reviewed the impact of the proposed modifications against the Salem setpoint study to verify that the accuracy of the temperature related functions are met. Salem presently assumes a 3.5% error in primary flow determination. This allowance continues to be conservative.

The failure of an RTD can be detected by the deviation alarm. This alarm is set for a measurement deviation when T_{ave} is calculated and also when delta T is calculated.

The impact of the RTD bypass elimination for the Salem plants on FSAR Chapter 15 non-LOCA accidents has been evaluated by the licensee. Since the effect of the temperature response time and accuracy of the new system is not degraded, the former conclusions in the FSAR remain valid.

The elimination of the RTD bypass system has been found to not impact the uncertainties associated with RCS temperature and flow measurement. It is concluded therefore that the elimination of the RTD bypass piping will not affect the LOCA analyses input and hence, the results of the analyses remain unaffected. Therefore, the plant design changes due to the RTD bypass elimination are acceptable from a LOCA analysis standpoint without requiring any reanalysis.

The staff's review and evaluation is also based upon Sections 7.2 and 7.3 of the Standard Review Plan. Those sections state that the objectives of the review are to confirm that the reactor trip and engineered safety features actuation system satisfy the requirements of the acceptance criteria and guidelines applicable to the protection system and will perform their safety function during all plant conditions for which they are required. Since our review indicates that the modified system does not functionally change (except three hot leg RTD's are utilized instead of just one) the reactor trip and engineered safety features actuation systems, the staff's original conclusions for these systems, as documented in Sections 7.2 and 7.3 of the SER dated October 11, 1974, for Salem Nuclear Generating Station remain valid. Based on this and the licensee's statement that the new hardware for the RTD bypass elimination has been qualified to IEEE Std 323-1974, IEEE Std 344-1975 and 10 CFR 50.49, we find the plant modifications to eliminate the RTD bypass manifold and to install fast response RTD's directly in the reactor coolant system hot and cold legs to be acceptable.

As a result of the new instrumentation associated with the removal of the existing RTD bypass manifold and replacement by dual element RTD's, the following changes to the plants' Technical Specifications were proposed:

CHANGE 1 - Change the entry under "ALLOWABLE VALUES" for Functional Unit 8, Overpower ΔT, in Table 2.2-1 from "See Note 3" to "See Note 4" for Salem Unit 1 and Unit 2.

CHANGE 2 - On page 2-9 add a new note 4, "The channel's maximum trip point shall not exceed its computed trip point by more than 3.0 percent," for both units covering a new allowable value for overpower ΔT.

CHANGE 3 - Change the allowable value (for overtemperature ΔT) in Note 3 to Table 2.2-1 from "4.0 percent" to "3.1 percent" for both units.

CHANGE 4 - Change the entry under "RESPONSE TIME" for Functional Unit 7, Overtemperature ΔT, in Table 3.3-2 from "4.0" to "5.75" for both units.

CHANGE 5 - Under "TABLE NOTATION" for Table 3.3-5, change identification of notes from symbols to numbers 1 thru 6 for Unit 1 only. In Table 3.3-5 change all references to notes correspondingly.

CHANGE 6 - Change the entry for the response time for Functional Units 2.f, 4.f, and 6.f, Auxiliary Feedwater Pumps, in Table 3.3-5 from "Not Applicable" to "60" for Unit 1 only.

CHANGE 7 - Increase all the response time entries by 1.75 seconds for Functional Unit 5, Steam Flow in two Steam Lines - High Coincident, in Table 3.3-5 for both units.

CHANGE 8 - Change the entry for the response time for Functional Unit 5.f, Auxiliary Feedwater Pumps, in Table 3.3-5 from "Not Applicable" to "61.75" for Unit 1 only.

CHANGE 9 - On the bottom of page 3/4 3-30 delete the note, "Response time for Motor-Driven Auxiliary Feedwater Pumps on all SI signal starts $\leq$60."

CHANGE 10 - Add a new Function Unit 14, Station Blackout, to Table 3.3-5 for Unit 2 only. Include a new response time entry for "Motor-Driven Auxiliary Feed Pumps" of "≦60."

Changes 1, 2, and 3 are necessary to reflect new allowable values based on revised instrumentation uncertainties resulting from the bypass manifold elimination. These new values were calculated using essentially the Westinghouse setpoint methodology as previously approved by the staff for generic use (see NUREG-0717, SER for Virgil C. Summer Nuclear Station) and are also more conservative. The staff finds these changes acceptable.

Changes 4 and 7 are new values based on revised individual component response times resulting from the bypass manifold removal. Since the new individual response times produce total response times for the reactor trips and engineered safety features actuation which remain the same as those used in approved safety analyses for Salem Unit 1 and Unit 2, we find these changes acceptable.

Change 5 is an editorial change intended to add agreement between the Technical Specifications for Unit 1 and Unit 2. On the basis that the change in purely editorial, we find it acceptable.

Changes 6 and 8 are new entries for Unit 1 which add conservatism and consistency with Unit 2 Technical Specifications. On this basis, we find them acceptable. Change 8 also reflects the revision resulting from Change 7 approved above.

Change 9 results from Changes 6 and 8 approved above. On the basis that the note to be deleted is no longer necessary because of the additional entries provided by Changes 6 and 8, the staff finds this change acceptable.

Change 10 provides a new entry for Unit 2 Technical Specifications which adds conservatism and consistency with Unit 1 Technical Specifications. On this basis, we find them acceptable.

The staff reviewed the licensee's intended inspections of field machined surfaces, welds, and weld materials to assure that all Code requirements will be met. The staff has concluded that all new hot and cold leg connections and penetrations, and crossover piping capping meet appropriate Code inspection requirements. In addition, both the hot and cold leg, the nozzle, thermowell, and the entire thermowell/nozzle assembly were analyzed to the ASME Code, Section III, Class I. The analysis of the entire assembly considered the weight of the RTD, the RTD head assembly and an assumed length of cabling. The effect of seismic and flow induced loads were considered. Therefore, the staff concludes that the analyses of the RCS penetrations are acceptable.

Finally, the licensee has provided adequate assurance that all significant radiological conditions have been considered by:

1. Identifying all major construction steps in the proposed RTD bypass system removal which could result in radiation exposure or generate radioactive wastes.

2. Providing a dose estimation for the RTD bypass system modifications, performing manpower projections and work time estimates for these work areas, performing dose estimates for major RTD bypass system removal subtasks and the overall task (77 person-rem per unit).

3. Projecting a net savings of 3,000 person-rem over plant life by RTD bypass system removal; assuming a 40 year operating license; and providing a comparison of dose incurred from task performance (77 person-rem per unit), and dose avoided through reduced maintenance and operational requirements (95 person-rem per unit). In addition, one outage day may be saved on each unit due to the avoidance of leaks and equipment failures.

4. Identifying specific ALARA measures to be employed for RTD bypass system removal, including preplanning of removal methods, use of temporary shielding, outage sequencing to minimize work area dose rates, work area familiarization; use of special tooling, and pre-job planning among job supervisors, health physics technicians, and ALARA staff.

5. Identifying the sources, types, volumes, and relative level of radioactive wastes which should result from RTD bypass system removal.

6. Evaluating the tasks for special radiological or operational considerations which could impact radiological conditions, or result in delays and additional exposures.

Additionally, the licensee's corporate and facility radiation protection and ALARA programs have previously been evaluated (i.e., in the Salem 1 and 2 SER, Chapter 12) and found to be adequate for radiological protection of workers, the general public, and the environment.

The staff has evaluated the radiological aspects of the RTD bypass system removal using the criteria of Chapter 12 of the Standard Review Plan (NUREG-0800), Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and licensee's commitments in the Salem Updated Final Safety Analysis Report (UFSAR - Revision 7, July 22, 1987), and concludes that the radiological aspects of the RTD bypass system removal have been fully considered and the radiation protection measures planned for the task are adequate to protect the worker, the general public and the environment; and will result in doses that are as low as is reasonably achievable (ALARA).

Based on the above the staff finds that the licensee's radiological protective measures can be expected to be conducted in accordance with the requirements of 10 CFR Part 20 and within ALARA guidelines, and are adequate for ensuring that occupational radiation exposures will be ALARA. We therefore find the radiation protection aspects of the RTD bypass system removal to support the Technical Specifications change acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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. 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.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 23106) on June 17, 1987 and consulted with the State of New Jersey. No public comments were received and the State of New Jersey did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributors: F. Burrows, H. Balukjian, J. L. Minns
D. C. Fischer

Dated: November 16, 1987