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OCT 0 8 1981

Docket No. 50-272

RECEIVED OCT 1 9 1981 U.S. NUCLEAR REGULATORY COMMISSION

Mr. F. W. Schneider, Vice President
Production
Public Service Electric and Gas Company
80 Park Plaza 15A
Newark, New Jersey 07101

Dear Mr. Schneider:

The Commission has issued the enclosed Amendment No. 39 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated September 29, 1981. The Technical Specifications changes are supported by the Safety Evaluation Report as transmitted to the Public Service Electric and Gas Company by letter dated March 21, 1981.

This amendment incorporates the requirements for implementation of the TMI-2 Lessons Learned Category "A" items. It includes the areas in the Safety Technical Specifications (Appendix A) of emergency power supply requirements, valve position indication, instrumentation for inadequate core cooling, containment isolation and auxiliary feedwater systems, and new license requirements for the implementation of programs to reduce leakage outside containment, to accurately determine airborne iodine concentration, and to ensure the capability to accurately monitor the Reactor Coolant System subcooling margin.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the

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proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 39 to DPR-70
2. Notice of Issuance

cc w/enclosures:
See next page

no legal objection

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

PUBLIC SERVICE ELECTRIC AND 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. 39
License No. DPR-70

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

   A. The application for amendment by Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated September 29, 1981, 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;

   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

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 39, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license is also amended by the addition of new paragraphs 2.C.(7), 2.C.(8) and 2.C.(9) that read as follows:

(7) Systems Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(8) Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel;

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

(9) Backup Method for Determining Subcooling Margin

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 8, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
IV	IV
X	X
---------	3/4 3-20a
3/4 3-22	3/4 3-22
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28
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---------	3/4 4-4a
3/4 4-5	3/4 4-5
3/4 4-6	3/4 4-6
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13

ATTACHMENT (CONTINUED)

Remove Pages	Insert Pages
3/4 6-17	3/4 6-17
3/4 7-5	3/4 7-5
3/4 7-6	3/4 7-6
B3/4 3-3	B3/4 3-3
B3/4 4-1a	B3/4 4-1a
B3/4 4-2	B3/4 4-2

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION | PAGE

3/4.3 INSTRUMENTATION

3/4.4 REACTOR COOLANT SYSTEM

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. AUXILIARY FEEDWATER					
a. Automatic Actuation Logic**	2	1	2	1, 2, 3	20
b. Stm. Gen. Water Level-Low-Low					
i. Start Motor Driven Pumps	3/stm. gen	2/stm. gen. any stm. gen.	2 stm. gen.	1, 2, 3	14*
ii. Start Turbine-Driven Pumps	3/stm. gen.	2/stm. gen. any 2 stm. gen.	2 stm. gen.	1, 2, 3	14*
c. Undervoltage-RCP Start Turbine-Driven Pump	4-1/bus	1/2 x 2	3	1, 2	19
d. S.I. Start Motor-Driven Pumps	See 1 above (All S.I. initiating functions and requirements)				
e. Emergency Trip of Steam Generator Feedwater Pumps - start Motor Driven Pumps	2-1/pump	2	2 -1/pump	1	21
f. Station Blackout	See 6 and 7 above (SEC and U/V Vital Bus)				

** Applies to items b. and c.

TABLE 3.3-3 (Continued)

ACTION 17 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ENGINEERED SAFETY FEATURES INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-11	With 2 of 3 pressurizer pressure channels ≥ 1925 psig.	P-11 prevents or defeats manual block of safety injection actuation on low pressurizer pressure.
P-12	With 3 of 4 T_{avg} channels ≥ 545°F.	P-12 prevents or defeats manual block of safety injection actuation high steam line flow and low steam line pressure.
	With 2 of 4 T_{avg} channels < 541°F.	Allows manual block of safety injection actuation on high steam line flow and low steam line pressure. Causes steam line isolation on high steam flow. Affects steam dump blocks.

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may proceed provided that either:

a. The inoperable channel is restored to OPERABLE within 72 hours, or

b. If the affected Steam Generator Feedwater Pump is expected to be out of service for more than 72 hours, the inoperable channel is jumpered so as to enable the Start Circuit of the Auxiliary Feedwater Pumps upon the loss of the other Steam Generator Feedwater Pump.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water Level--High-High	$<$ 67% of narrow range instrument span each steam generator	$<$ 68% of narrow range instrument span each steam generator
6. SAFEGUARDS EQUIPMENT CONTROL SYSTEM (SEC)	Not Applicable	Not Applicable
7. UNDERVOLTAGE, VITAL BUS		
a. Loss of Voltage	$\geq$ 70%	$\geq$ 65%
8. AUXILIARY FEEDWATER		
a. Automatic Actuation Logic	Not Applicable	Not Applicable
b. Steam Generator Water Level-low-low	$\geq$ 18% of narrow range instrument span each steam generator	$\geq$ 17% of narrow range instrument span each steam generator
c. Undervoltage - RCP	$\geq$ 70% RCP bus voltage	$\geq$ 65% RCP bus voltage
d. S.I.	See 1 Above (All S.I. setpoints)	
e. Emergency Trip of Steam Generator Feedwater Pumps	Not Applicable	Not Applicable
f. Station Blackout	See 6 and 7 above (SEC and Undervoltage, Vital Bus)	

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*
b. Reactor Trip (from SI)	$\leq$ 2.0
c. Feedwater Isolation	$\leq$ 7.0
d. Containment Isolation-Phase "A"	$\leq$ 17.0#/27.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	$\leq$ 13.0#/48.0**

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27.0*/12.0#
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
d. Containment Isolation-Phase "A"	≤ 18.0#
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	≤49.0*/13.0#
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	≤12.0#/22.0#
b. Reactor Trip (from SI)	≤2.0
c. Feedwater Isolation	≤7.0
d. Containment Isolation-Phase "A"	≤17.0#/27.0#
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	≤ 13.0#/48.0##
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	≤14.0#/24.0##
b. Reactor Trip (from SI)	≤4.0
c. Feedwater Isolation	≤9.0
d. Containment Isolation-Phase "A"	≤ 19.0#/29.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	≤ 14.0#/49.0##
h. Steam Line Isolation	≤ 9.0

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#/22.0##
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 7.0
d. Containment Isolation-Phase "A"	≤ 17.0#/27.0##
e. Containment Ventilation Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Service Water System	≤ 14.0#/48.0**
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-Reactor Trip	≤ 2.5
b. Feedwater Isolation	≤ 11.0
9. Steam Generator Water Level --Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine-Driven Auxiliary Feedwater Pumps	≤ 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. Containment Pressure-Vacuum Relief System Isolation	$\leq$ 5.0 (***)
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 Feedwater Pumps	$\leq$ 60.0

Note: Response time for Motor-driven Auxiliary Feedwater Pumps on all S.I. signal starts $\leq$ 60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

(*)) 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.

(#)) 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.

(##)) 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.

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

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

(***)) The response time is the time the isolation circuitry input reaches the isolation setpoint to the time theContainment Pressure-Vacuum Relief valves are fully shut.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONTAINMENT ISOLATION				
a. Phase "A" Isolation				
1) Manual	N.A.	N.A.	R	1, 2, 3, 4
2) From Safety Injection Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
b. Phase "B" Isolation				
1) Manual	N.A.	N.A.	R	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Pressure--High-High	S	R	M(3)	1, 2, 3
c. Containment Ventilation Isolation				
1) Manual	N.A.	N.A.	R	1, 2, 3, 4
2) Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
3) Containment Radio-activity-High	S	R	M	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
8.	AUXILIARY FEEDWATER				
a.	Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
b.	Steam Generator Water Level-Low-Low	S	R	M	1, 2, 3
c.	Undervoltage - RCP	S	R	M (2)	1, 2
d.	S.I.	See 1 above (All S.I. surveillance requirements)			
e.	Emergency Trip of Steam Generator Feedwater Pumps	N.A.	N.A.	R	1
f.	Station Blackout	See 6b and 7 above (SEC and U/V Vital Bus)			

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

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

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

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11a and Table 3.3-11b shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. As shown in Table 3.3-11a and Table 3.3-11b.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-11.

TABLE 3.3-11a

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	REQUIRED NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	4 (1/loop)	2	1
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	4 (1/loop)	2	1
3. Reactor Coolant Pressure (Wide Range)	2	2	1
4. Pressurizer Water Level	3 (hot)	2	1
5. Steam Line Pressure	3/Steam Generator	2/Steam Generator	1
6. Steam Generator Water Level (Narrow Range)	3/Steam Generator	2/Steam Generator	1
7. Steam Generator Water Level (Wide Range)	4 (1/Steam Generator)	4 (1/Steam Generator)	1
8. Refueling Water Storage Tank Water Level	2	2	1
9. Boric Acid Tank Solution Level	2 (1/tank)	2 (1/tank)	3
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	4 (1/Steam Generator)	4
11. Reactor Coolant System Subcooling Margin Monitor	2*	2*	5
12. PORV Position Indicator	2/valve**	2/valve**	1
13. PORV Block Valve Position Indicator	2/valve**	2/valve**	1
14. Pressurizer Safety Valve Position Indicator	2/valve**	2/valve**	1

(*) Total number of channels is considered to be two (2) with one (1) of the channels being manual calculation by licensed control room personnel using data from OPERABLE wide range Reactor Coolant Pressure and Temperature along with Steam Tables as described in ACTION 5.

(**) Total number of Channels is considered to be two (2) with one (1) of the channels being any one (1) of the following alternate means of determining PORV, PORV Block, or Safety Valve position: Tailpipe Temperatures for the valves, Pressurizer Relief Tank Temperature, Pressurizer Relief Tank Level OPERABLE.

TABLE 3.3-11b

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	4 (1/loop)	1	2
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	4 (1/loop)	1	2
3. Reactor Coolant Pressure (Wide Range)	2	1	2
4. Pressurizer Water Level	3 (hot)	1	2
5. Steam Line Pressure	3/Steam Generator	1/Steam Generator	2
6. Steam Generator Water Level (Narrow Range)	3/Steam Generator	1/Steam Generator	2
7. Steam Generator Water Level (Wide Range)	4 (1/Steam Generator)	3 (1/Steam Generator)	2
8. Refueling Water Storage Tank Water Level	2	1	2
9. Boric Acid Tank Solution Level	2 (1/tank)	1	2
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	3 (1/Steam Generator)	6
11. Reactor Coolant System Subcooling Margin Monitor	2*	1	6
12. PORV Position Indicator	2/valve**	1	2
13. PORV Block Valve Position Indicator	2/valve**	1	2
14. Pressurizer Safety Valve Position Indicator	2/valve**	1	2

(*) Total number of channels is considered to be two (2) with one (1) of the channels being manual calculation by licensed control room personnel using data from OPERABLE wide range Reactor Coolant Pressure and Temperature along with Steam Tables as described in ACTION 5.

(**) Total number of Channels is considered to be two (2) with one (1) of the channels being any one (1) of the following alternate means of determining PORV, PORV Block, or Safety Valve position: Tailpipe Temperatures for the valves, Pressurizer Relief Tank Temperature, Pressurizer Relief Tank Level OPERABLE.

TABLE 3.3-11a&b (continued)

TABLE NOTATION

ACTION 1 — With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11a, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 2 — With the number of OPERABLE accident monitoring channels less than the Minimum Number of Channels shown in Table 3.3-11b, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

ACTION 3 — With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11a, operation may proceed provided that the Boric Acid Tank associated with the remaining OPERABLE channel satisfies all requirements of Specification 3.1.2.8.a.

ACTION 4 — With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11a, operation may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate channel.

ACTION 5 — With the number of OPERABLE channels less than the Required Number of Channels shown in Table 3.3-11a, operation may proceed provided that Steam Tables are available in the Control Room and the following Required Channels shown in Table 3.3-11a are OPERABLE to provide an alternate means of calculating Reactor Coolant System subcooling margin:

a. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)

b. Reactor Coolant Pressure (Wide Range)

ACTION 6 — With the number of OPERABLE channels less than the Minimum Number of Channels shown in Table 3.3-11b, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

TABLE 4.3-11

SURVEILLANCE REQUIREMENTS FOR ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R	NA
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R	NA
3. Reactor Coolant Pressure (Wide Range)	M	R	NA
4. Pressurizer Water Level	M	R	NA
5. Steam Line Pressure	M	R	NA
6. Steam Generator Water Level (Narrow Range)	M	R	NA
7. Steam Generator Water Level (Wide Range)	M	R	NA
8. Refueling Water Storage Tank Water Level	M	R	NA
9. Boric Acid Tank Solution Level	M	R	NA
10. Auxiliary Feedwater Flow Rate	SU#	R	NA
11. Reactor Coolant System Subcooling Margin Monitor	M	NA*	NA
12. PORV Position Indicator	M	NA	Q
13. PORV Block Valve Position Indicator	M	NA	Q
14. Pressurizer Safety Valve Position Indicator	M	NA	R

Auxiliary Feedwater System is used on each Startup and Flow Rate indication is verified at that time.

* The instruments used to develop RCS Subcooling Margin are calibrated on an 18 Month cycle; the Monitor will be compared Quarterly with calculated subcooling margin for known input values.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.2 SAFETY VALVES

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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 more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION and operating the valve through one complete cycle of full travel.

4.4.3.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

REACTOR COOLANT SYSTEM

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1650 cubic feet (92% indicated level), and at least two groups of pressurizer heaters each having a capacity of 150 kw.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.4.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring circuit current at least once per 92 days.

4.4.4.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Verifying that on a feedwater isolation test signal, each feedwater isolation valve actuates to its isolation position.

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each Purge and Pressure-Vacuum Relief valve actuates to its isolation position.

4.6.3.1.3 At least once per 18 month, verify that on a main steam isolation test signal, each main steam isolation valve specified in Table 3.6-1 actuates to its isolation position.

4.6.3.1.4 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.1.5 Each containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then at least once per 72 hours, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to $0.60L_a$.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME (Seconds)
F. MANUAL		
1. 1SS900#	Pressurizer Dead-Weight Calibrator	Not Applicable
2. 1SS901#	Pressurizer Dead-Weight Calibrator	Not Applicable
3. 11 CV 98#	CVCS - RCP Seals	Not Applicable
4. 12 CV 98#	CVCS - RCP Seals	Not Applicable
5. 13 CV 98#	CVCS - RCP Seals	Not Applicable
6. 14 CV 98#	CVCS - RCP Seals	Not Applicable
7. 1 SJ 71#	CVCS Flushing Connection	Not Applicable
8. 11 SS 93*#	Steam Generator Sampling	Not Applicable
9. 12 SS 93*#	Steam Generator Sampling	Not Applicable
10. 13 SS 93*#	Steam Generator Sampling	Not Applicable
11. 14 SS 93*#	Steam Generator Sampling	Not Applicable
12. 1 SA 118#	Compressed Air Supply	Not Applicable
13. 1 WL 190#	Refueling Canal Supply	Not Applicable
14. 1 SF 36#	Refueling Canal Supply	Not Applicable
15. 1 WL 191#	Refueling Canal Discharge	Not Applicable
16. 1 SF 22#	Refueling Canal Discharge	Not Applicable
17. 1 VC 9*#	Containment Radiation Sampling	Not Applicable
18. 1 VC 10*#	Containment Radiation Sampling	Not Applicable
19. 1 VC 13*#	Containment Radiation Sampling	Not Applicable
20. 1 VC 14*#	Containment Radiation Sampling	Not Applicable
21. - #	Fuel Transfer Tube	Not Applicable

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated manual activation switches in the control room and flow paths shall be OPERABLE with:

a. Two feedwater pumps, each capable of being powered from separate vital busses, and

b. One feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps 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.

b. With two auxiliary feedwater pumps inoperable be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1275 psig on recirculation flow.

2. Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1500 psig on recirculation flow when the secondary steam supply pressure is greater than 750 psig. The provisions of Specification 4.0.4 are not applicable.

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verify that valves 11AF3, 12AF3, 13AF3, 11AF20, 12AF20, 13AF20, 14AF20, 11AF22, 12AF22, 13AF22, 14AF22, 11AF10, 12AF10, 13AF10, 14AF10, 11AF86, 12AF86, 13AF86, and 14AF86 are locked open.

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the motor driven pump flow path actuates to its correct position on a pump discharge pressure test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

c. The auxiliary feedwater system shall be demonstrated OPERABLE prior to entry into Mode 3 following each COLD SHUTDOWN by performing a flow test to verify the normal flow paths from the Auxiliary Feedwater Storage Tank to each of the steam generaters.

3/4.3.3.6 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the Recommendations of Regulator Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following and Accident," December 1975.

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 lbs per hour of saturated steam at the valve set point. 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.

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 set point 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 power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each power operated relief valve has a remotely operated block valve to provide positive shutoff capability should a relief valve become inoperable.

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 assures that the plant will be able to establish natural circulation.

3/4.4.5 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.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-272

PUBLIC SERVICE ELECTRIC AND GAS COMPANY,
PHILADELPHIA ELECTRIC COMPANY,
DELMARVA POWER AND LIGHT COMPANY, AND
ATLANTIC CITY ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 39 to Facility Operating License No. DPR-70, issued to Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees), which revised the Facility Operating License and Technical Specifications for operation of the Salem Nuclear Generating Station, Unit No. 1 (the facility) located in Salem County, New Jersey. The amendment is effective as of the date of issuance.

The amendment incorporates the requirements for implementation of the TMI-2 Lessons Learned Category "A" items. It includes the areas in the Safety Technical Specifications (Appendix A) of emergency power supply requirements, valve position indication, instrumentation for inadequate core cooling, containment isolation and auxiliary feedwater systems; and adds new license requirements for the implementation of programs to reduce leakage outside containment to accurately determine airborne iodine concentration, and to ensure the capability to accurately monitor the Reactor Coolant System subcooling margin.

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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated September 29, 1981, (2) Amendment No. 39 to License No. DPR-70, (3) the Commission's letter dated October 8, 1981, and (4) the Commission's related Safety Evaluation dated March 21, 1981. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Salem Free Public Library, 112 West Broadway, Salem, New Jersey. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 8th day of October, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing