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Docket No. 50-272

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Public Service Electric & Gas Company
ATTN: Mr. F. P. Librizzi
General Manager - Electric Production
Production Department
80 Park Place, Room 7221
Newark, New Jersey 07101

Gentlemen:

The Commission has issued the enclosed Amendment No. 9 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 applications dated August 19, September 20, September 26, October 7, October 28, November 17 and November 17, 1977.

The amendment to the Technical Specifications will (1) revise the section dealing with routine reports and reportable occurrences to be consistent with a recent change to Commission's guidance, (2) change the title of one member of the Nuclear Review Board and designate a different Vice Chairman in order to maintain the same professional qualifications of the Board as originally reviewed and approved by the Commission, (3) revise the pressurizer heatup rate to be consistent with design limits, and (4) make certain editorial corrections to rectify errors contained in the Technical Specifications as originally issued with DPR-70.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by
D. M. Verrelli for

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 9
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosure:
See next page

OFFICE➤	ORB#3	ORB#3	STSG	OELD	ORB#3
SURNAME➤	CParrish	DVerrelli:ar	JMcGough	B. Smith	GLear
DATE➤	12/13/77	12/13/77	12/13/77	12/23/77	12/27/77

NRC FORM 318 (9-76) NRCM 0240

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cc:

Richard Fryling, Jr., Esquire
Assistant General Counsel
Public Service Electric & Gas Company
80 Park Place
Newark, New Jersey 07101

Troy B. Conner, Jr., Esquire
Suite 1050
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

Gene Fisher
Bureau Chief
Bureau of Radiation Protection
380 Scotts Road
Trenton, New Jersey 08628

Honorable Samuel Donolson
Mayor, Lower Alloways Creek Township
Salem County, New Jersey 08079

State House Annex
ATTN: Deputy Attorney General
State of New Jersey
36 West State Street
Trenton, New Jersey 08625

Attorney General
Department of Law & Public Safety
State House Annex
Trenton, New Jersey 08625

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

Public Service Electric & Gas Company
ATTN: Herbert J. Heller
Manager, Salem Nuclear Generating Station
Hancocks Bridge, New Jersey 08038

Chief, Energy Systems Analysis Br. (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Salem Free Library
112 West Broadway
Salem, New Jersey 08079

Public Service Electric & Gas Co.
ATTN: R. L. Mittl
General Manager - Licensing and Environment
80 Park Place
Newark, New Jersey 07101

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. 9
License No. DPR-70

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

   A. The applications 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 August 19, September 20, September 26, October 7, November 17 and November 17, 1977, comply 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. 9, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David M. Verrelli

for George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 27, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 9

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

XVI
1-1
1-4
3/4 1-25
3/4 2-7
3/4 2-11
3/4 3-47
3/4 4-29
B 3/4 2-5
6-8
6-13
6-13a (added)
6-13b (added)
6-13c (added)
6-13d (added)

ADMINISTRATIVE CONTROLS

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3338 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part length rod position, and

b. All full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow from the reactor coolant pump seals.

QUADRANT POWER TILT RATIO

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

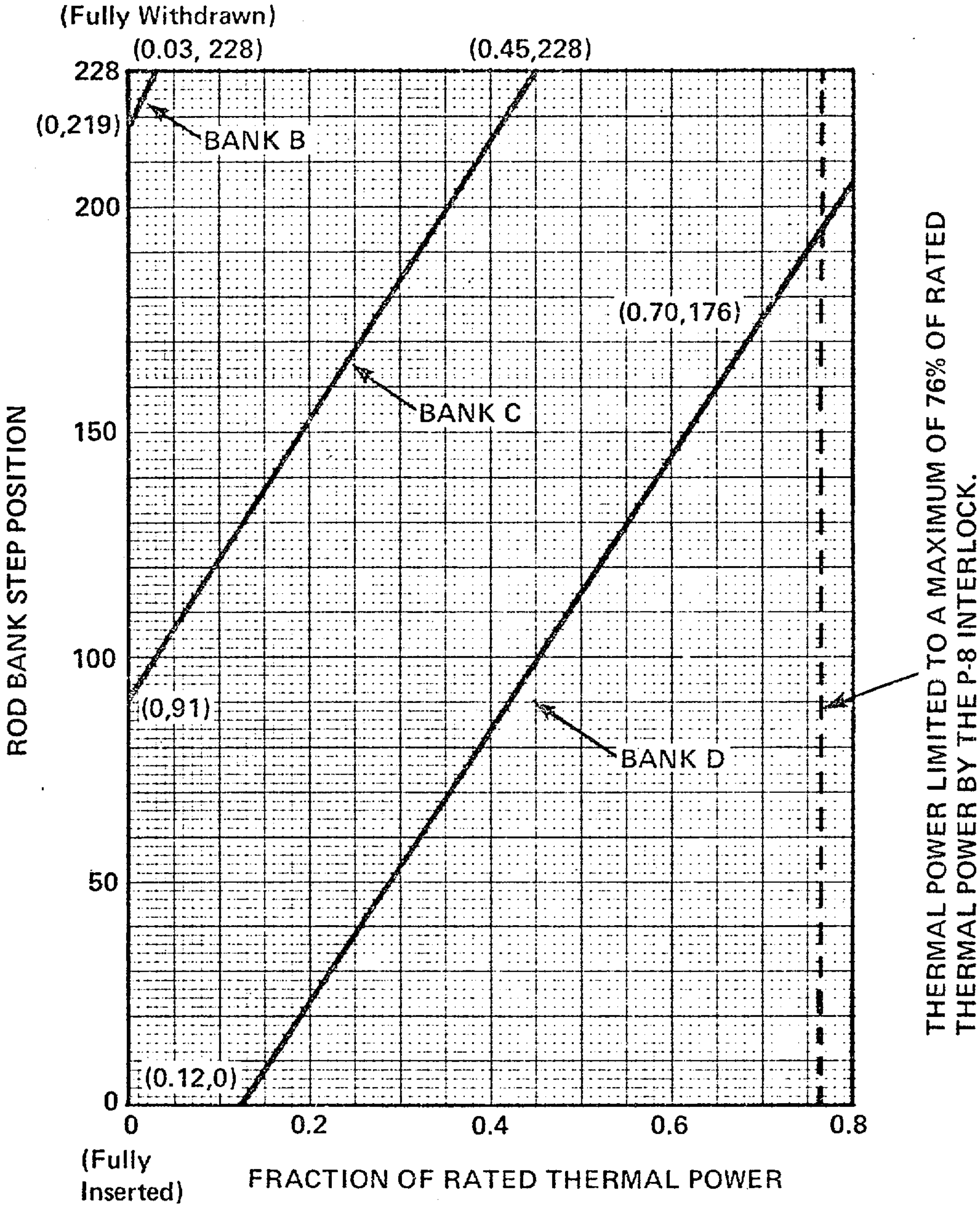

FIGURE 3.1-2 ROD BANK INSERTION LIMITS VERSUS THERMAL POWER THREE LOOP OPERATION

PART LENGTH ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 All part length rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*

ACTION:

With a maximum of one part length rod not fully withdrawn, within one hour either:

a. Fully withdraw the rod, or

b. Be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 Each part length rod shall be determined to be fully withdrawn by:

a. Verifying the position of the part length rod prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER, and

b. Verifying, at least once per 31 days, that electric power has been disconnected from its drive mechanism by physical removal of a breaker from the circuit.

---

*See Special Test Exceptions 3.10.2 and 3.10.3

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing either bank "D" control rods or any part length rods, and

2. $F_{xy}^{RTP} \leq 1.55$ for all unrodded core planes.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 $\pm$ 2%, 32.1 $\pm$ 2%, 46.4 $\pm$ 2%, 60.6 $\pm$ 2% and 74.9 $\pm$ 2%, inclusive.
4. Core plane regions within $\pm$ 2% of core height ($\pm$ 2.88 inches) about the bank demand position of the bank "D" or part length control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever F_{xy}^{C} exceeds F_{xy}^{L}.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

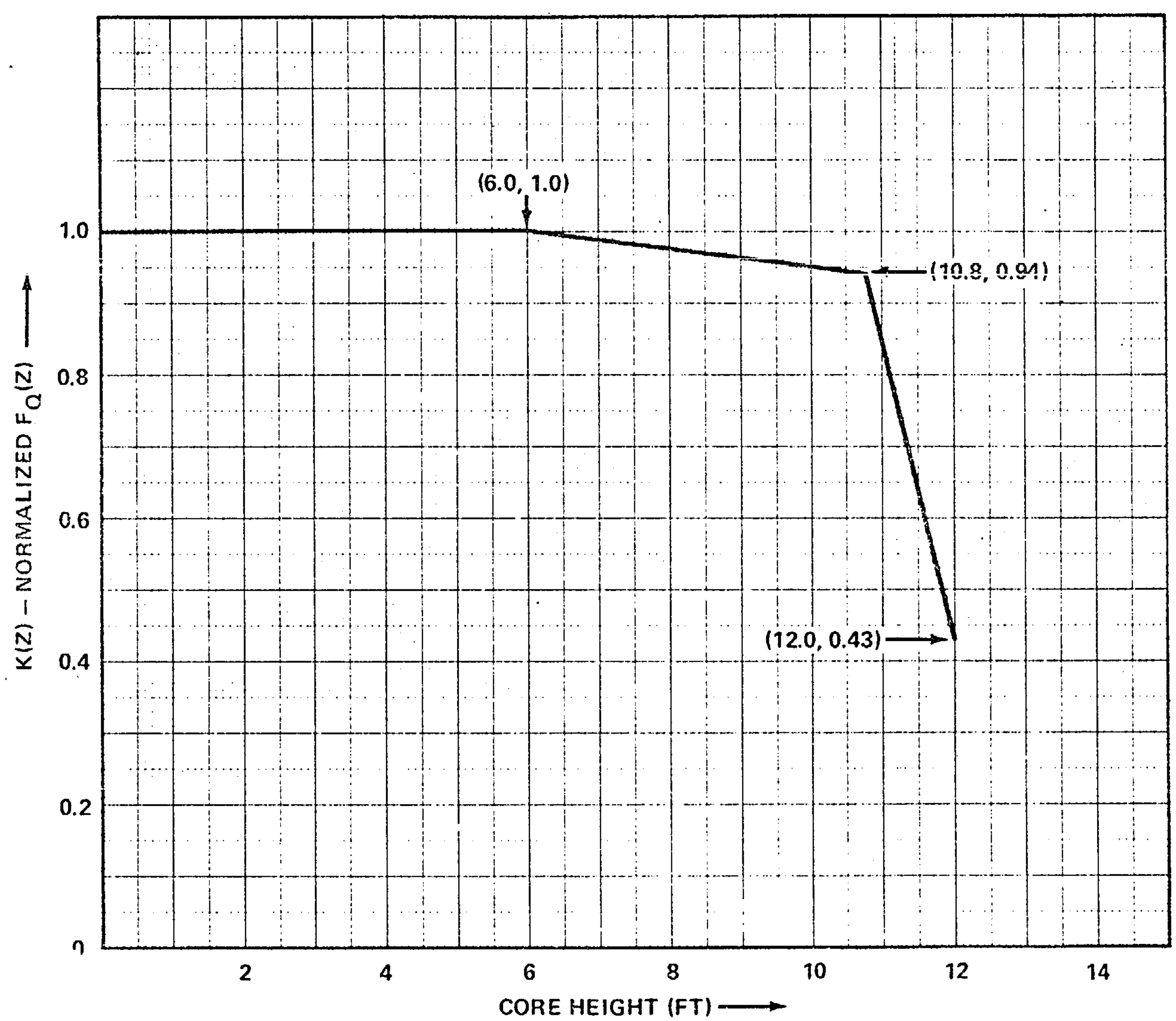

FIGURE 3.2-2

K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but $\leq$ 1.09:

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to $\leq$ 55% of RATED THERMAL POWER within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown, control or part length rod:

   1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown, control or part length rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

c. Using the movable incore detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Pressurizer Pressure	Hot Shutdown Panel 213	1700-2500 psig	1
2. Pressurizer Level	Hot Shutdown Panel 213	0 - 100%	1
3. Steam Generator Pressure	Hot Shutdown Panel 213	0 - 1200 psig	1/steam generator
4. Steam Generator Level	Hot Shutdown Panel 213	0 - 100%	1/steam generator

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Pressurizer Pressure	M	R
2. Pressurizer Level	M	R
3. Steam Generator Pressure	M	R
4. Steam Generator Level	M	R

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in F_Q by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3 percent from RATED THERMAL POWER for each percent of tilt in excess of 1.0.

BASES

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear Review Board.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear Review Board.

AUTHORITY

6.5.1.7 The Station Operations Review Committee shall:

a. Recommend to the Station Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the General Manager-Electric Production and the Nuclear Review Board of disagreement between the SORC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Station Operations Review Committee shall maintain written minutes of each meeting and copies shall be provided to the General Manager-Electric Production and Chairman of the Nuclear Review Board.

6.5.2 NUCLEAR REVIEW BOARD (NRB)

FUNCTION

6.5.2.1 The Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The NRB shall be composed of the:

Chairman:	General Manager-Electric Production
Vice Chairman:	Assistant to General Manager-Fuel Supply
Member:	General Manager-Licensing and Environment
Member:	Manager-Nuclear Operations
Member:	Manager-Quality Assurance
Member:	Project Manager-Hope Creek
Member:	Manager-Salem Generating Station
Member:	Principal Staff Engineer

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NRB Chairman to provide expert advice to the NRB.

MEETING FREQUENCY

6.5.2.5 The NRB shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the SORC and approved by the Station Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparision of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of OI&E, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsmile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfilment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

6.9 REPORTING REQUIREMENTS (Continued)

d. Seismic event analysis, Specification 4.3.3.3.2.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source and fission detector leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 9 TO FACILITY OPERATING LICENSE DPR-70

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

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. 50-272

Introduction

By letters dated August 19, 1977, September 20, 1977, September 26, 1977 October 7, 1977, October 28, 1977, November 17, 1977 and November 17, 1977, Public Service Electric and Gas Company (PSE&G) proposed changes to the Technical Specifications appended to Operating License No. DPR-70 for the Salem Nuclear Generating Station Unit No. 1. The proposed changes would (1) revise the section dealing with routine reports and reportable occurrences to be consistent with a recent change to Commission guidance, (2) change the title of one member of the Nuclear Review Board and designate a different Vice Chairman in order to maintain the same professional qualifications of the Board as originally reviewed and approved by the Commission, (3) revise the pressurizer heatup rate to be consistent with design limits, and (4) make certain editorial corrections to rectify errors contained in the Specifications as originally issued with DPR-70.

Background

With respect to the PSE&G proposed revision to routine reporting requirements, the background information influencing their submittal is summarized below.

Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", is the basis for reporting requirements found in Technical Specifications today. When these Technical Specifications were issued we requested that licensees use the formats in the guide for the Licensee Event Report (LER) and Monthly Operating Report. In some cases licensees' use of these formats was required by a reference to Regulatory Guide 1.16 in the Technical Specifications. After two years of experience with the reporting requirements identified in this guide we reviewed the scope of information licensees are required to submit in the LER, Annual Operating Report, Monthly Operating Report and Startup Report.

Based on our review of LER's we developed a modified format for the LER to make this document more useful for evaluation purposes. By letters sent in July and August 1977, we informed licensees of the new LER format and requested that they use it. For those licensees who reference Regulatory Guide 1.16 in their Technical Specifications we also requested that they propose a change which would replace this reference with appropriate words from the guide and which would delete mandatory use of the reporting forms contained in the guide.

From our review of all licensee reports we determined that much of the information found in the Annual Operating Report either is addressed in the LER's or Monthly Operating Reports, which are submitted in a more timely manner, or could be included in these reports with only a slight augmentation of the information already supplied. Therefore we concluded that the Annual Operating Report could be deleted as a Technical Specification requirement if certain additional information were provided in the Monthly Operating Reports. As a result we sent letters during September 1977 to licensees informing them that a revised and improved format for Monthly Operating Reports was available and requested that they use it. For those licensees with the Technical Specification reference to Regulatory Guide 1.16 the change deleting this reference, discussed above, would be necessary. In addition, licensees were informed that if they agreed to use the revised format they should submit a change request to delete the requirement for an Annual Operating Report except that occupational exposure data must still be submitted.

The PSE&G submittals of October 7, 1977 and October 28, 1977 are in response to our request of July 29, 1977 and September 22, 1977. The PSE&G submittals conform to the model specifications furnished by the staff.

Evaluation

1. Routine Reporting Requirements

The proposed change which would replace the reference to Regulatory Guide 1.16 with appropriate wording from that guide is administrative in nature and does not change the operation of the reactor. This change provides wording in the Technical Specifications which identifies the required reports, states the circumstances under which they should be submitted and details the timing of such submittals. The text does not specify in great detail the format and content of the reports as was previously done by reference to the guide. The proposed change provides greater flexibility to accommodate changes to the reporting system and allows the licensee to use the recently revised LER and Monthly Operating Report formats and is, therefore acceptable.

The licensee has also proposed to delete all but one of the four specified items in the Annual Operating Report. The report which tabulates occupational exposure on an annual basis is needed and, therefore, the requirement to submit this information has been retained. We have determined that the failed fuel examination information does not need to be supplied routinely by licensees because this type of historical data can be obtained in a compiled form from fuel vendors when needed. The information concerning forced reductions in power and outages will be supplied in the revised Monthly Operating Reports and the narrative summary of operating experience will be provided on a monthly basis in the Monthly Operating Report rather than annually. The licensee has committed to use the revised Monthly Operating Report format beginning with their report for January 1978 as requested. We have concluded that all needed information will be provided and deletion of the Annual Operating Report is acceptable.

2. NRB Membership

The licensee indicated that the currently designated Vice Chairman of the NRB has resigned from the Board, and the licensee proposed to revise the membership by replacing the General Manager - Construction with Principal Staff Engineer (a Company unique position) who has the same expertise (technical discipline) as the former individual. The staff has determined that since the qualifications and functions of the Board are not altered, the proposed change is acceptable.

3. Pressurizer Heatup Rate

Recently, it has come to the attention of the staff and of licensees with Westinghouse designed reactors that the pressurizer limiting heatup rate of 200°F/hr. is not consistent with design limits and should be changed to 100°F/hr. Discussion between the staff and Westinghouse resulted in the conclusion that the correction to this limit applies only to the heatup rate and that the present limit on cooldown rate of 200°F/hr. is acceptable. By letter dated October 13, 1977, we requested PSE&G to submit a change to the technical specifications to correct this error. PSE&G response of November 17, 1977, not only requested this change but also indicated that a survey of their records indicated that the pressurizer heatup rate at Salem Unit No. 1 has never exceeded 100°F/hr. Since the requested change to correct this error is more conservative than previously specified and is consistent with staff guidance as issued to the licensee, we find it acceptable as proposed.

4. Other Changes

The other changes proposed by the licensee are editorial in nature. The definition of controlled leakage has been changed to accurately reflect that seal water flow is measured from and not to the reactor coolant pump seals. The action statements associated with quadrant power tilt ratio have been clarified to indicate specifically that the out of limits penalty is a reduction of 3% from rated thermal power for each percent of tilt. Typographical errors have been discovered in the tabulation of the steam generator pressure instrumentation range and in the surveillance discription for measuring the heat flux hot channel factor. The figure in the current technical specifications incorrectly depicts the rod insertion limits for 3-loop operation and has been revised. Since none of these changes involve revision to operating limits or surveillance requirements, we have determined that they are acceptable as proposed.

Environmental Consideration

We have determined that this 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.

Conclusion

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 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.

Dated: December 27, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-272

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY
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. 9 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 operating license for Salem Nuclear Generating Station, Unit No. 1 (the facility) located in Salem County, New Jersey. The amendment is effective as of its date of issuance.

The amendment will (1) revise the section dealing with routine reports and reportable occurrences to be consistent with a recent change to Commission guidance, (2) change the title of one member of the Nuclear Review Board and designate a different Vice Chairman in order to maintain the same professional qualifications of the Board as originally reviewed and approved by the Commission, (3) revise the pressurizer heatup rate to be consistent with design limits, and (4) make certain editorial corrections to rectify errors contained in the Specifications as originally issued with DPR-70.

The applications for the amendment comply 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 applications for amendment dated October 19, September 20, September 26, October 7, October 28, November 17 and November 17, 1977, (2) Amendment No. 9 to License No. DPR-70 and (3) the Commission's related Safety Evaluation. 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) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 27 day of December 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

David M. Verrelli

David M. Verrelli, Acting Chief
Operating Reactors Branch #3
Division of Operating Reactors