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March 22, 1989

DO NOT REMOVE

Docket Nos. 50-272/311

POSTED
50-272
SALEM 1
AMENDMENT NO. 091
TO DPR-70

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

Dear Mr. Miltenberger:

SUBJECT: REDEFINITION OF FULLY WITHDRAWN (TAC NOS. 71834/71835)

RE: SALEM GENERATING STATION, UNIT NOS. 1 AND 2

The Commission has issued the enclosed Amendment Nos. 91 and 66 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 3, 1989 and supplemented by letter dated February 16, 1989. The supplemental letter provided additional clarifying information but did not change the original application.

These amendments redefine the FULLY WITHDRAWN position of the control rod cluster assemblies from 228 steps to a band between 222 and 228 steps withdrawn. Other changes: delete Figure 3.1.2 from Unit 1 TSs; delete Specification 3.10.5 from Unit 2 TSs; and incorporate rod testing requirements into Specification 3.1.3.2.2 for Unit 1 and Unit 2 TSs.

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

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

Enclosures:
1. Amendment No. 91 to License No. DPR-70
2. Amendment No. 61 to License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

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Docket File	MO'Brien (2)	Wanda Jones	Brent Clayton	
NRC PDR	OGC	EButcher	EWenzinger	
Local PDR	DHagan	Tech Branch	MHodges	
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BGrimes	CMiles, GPA/PA	JStone/MThadani	TMeek (8)	RDiggs, ARM/LFMB

[MILTENBERGER]

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D	SRXB
MO'Brien	JStone:tr	NDR	WButler	MHodges
3/ /89	3/6/89	3/14/89	3/22/89	3/6/89

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

March 22, 1989

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   License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

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

Salem Nuclear Generating Station

cc:

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 91
License No. DPR-70

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

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated January 3, 1989 and supplemented by letter dated February 16, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 91 , 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 startup from the eighth refueling outage, currently scheduled to begin April 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 22, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 91

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-3	1-3
1-6	1-6
B 2-2	B 2-2
3/4 1-20	3/4 1-20
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 1-24	3/4 1-24
3/4 1-25	3/4 1-25
3/4 10-1	3/4 10-1
B 3/4 1-4	B 3/4 1-4

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."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 228 steps withdrawn, inclusive. FULLY WITHDRAWN will be specified in the current reload analysis.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 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

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE OCCURRENCE

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

SHUTDOWN MARGIN

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

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee, as shown in Figure 5.1-3, and which defines the exclusion area as shown in Figure 5.1-1.

SOLIDIFICATION

1.30 SOLIDIFICATION shall be the conversion of wet radioactive wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.31 SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.32 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, BASES

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3(1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods FULLY WITHDRAWN the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1 1955 Edition while the valves are designed to ANSI B 16.5, MSS-SP-66-1964, or ASME Section III-1968, which permit maximum transient pressures of up to 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

POSITION INDICATION SYSTEM SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.2.2 The group demand position indicator shall be OPERABLE for each shutdown and control rod not fully inserted. During the performance of individual full length (shutdown and control) rod testing measurement during rod position indication system calibration:

a. Only one shutdown or control bank shall be withdrawn from the fully inserted position at a time, and

b. K_{eff} shall be maintained less than or equal to 0.95.

APPLICABILITY: MODES 3*, 4*, and 5*

ACTION:

With less than the above required group demand position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.2.2 Each of the above required group demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days.

*With the reactor trip system breakers in the closed position

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 228 steps withdrawn position shall be ≤ 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} ≥ 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to ≤71% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

POSITION INDICATION SYSTEM SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

APPLICABILITY: MODES 1*, and 2*#

ACTION:

With a maximum of one shutdown rod not FULLY WITHDRAWN, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. FULLY WITHDRAW the rod, or,

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators within one hour after rod motion:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor critically, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3

#With K_{eff} greater than or equal to 1.0

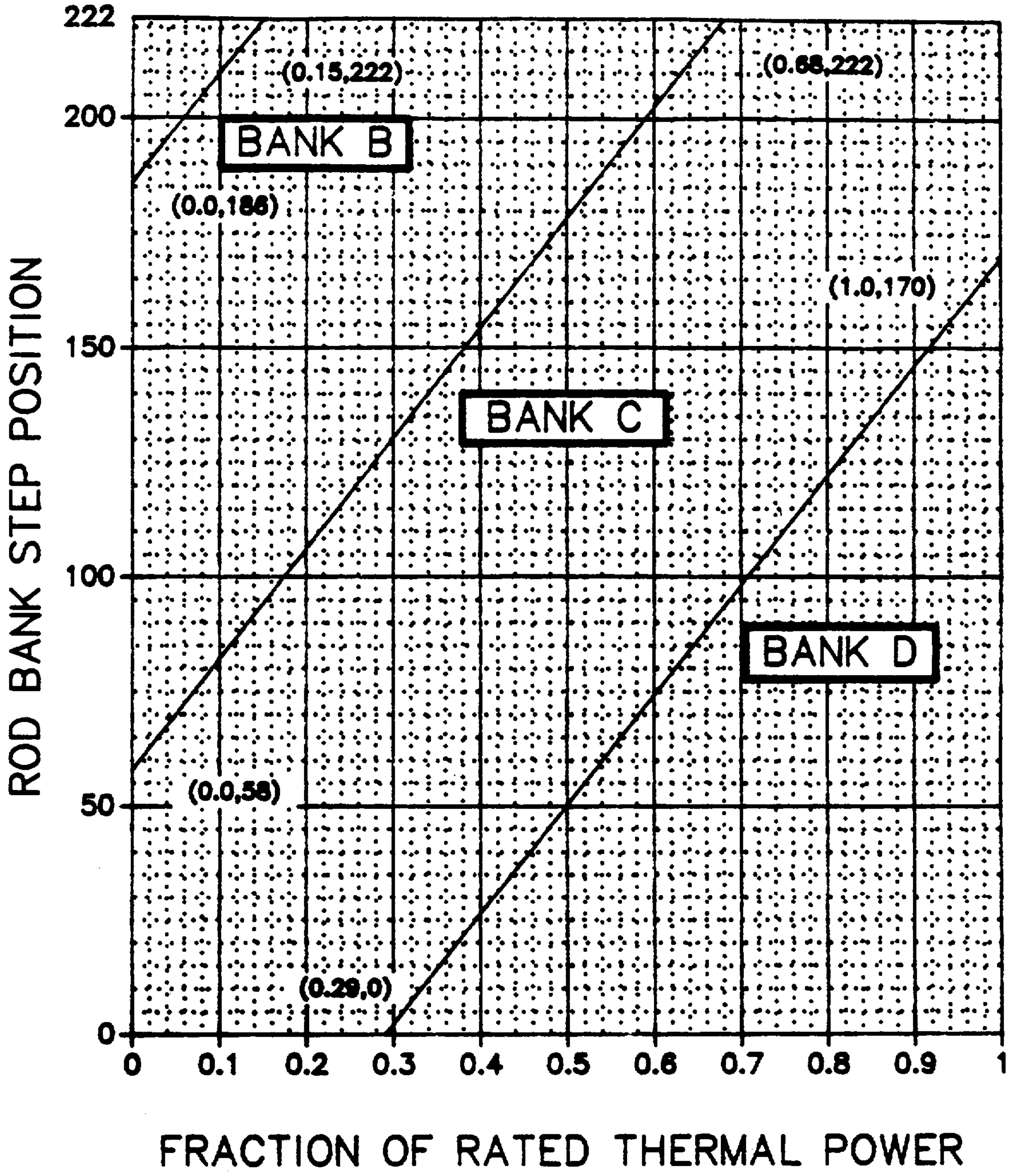

Figure 3.1-1
ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

FIGURE 3.1-2

<u>INTENTIONALLY LEFT BLANK PENDING</u>
<u>COMMISSION APPROVAL OF THREE LOOP OPERATION</u>

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

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s), and

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 10 gpm of 20,000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 10 gpm of 20,000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length rod either partially or FULLY WITHDRAWN shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod mis-alignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within ± 12 steps of the bank demand position for a range of positions. For the Shutdown Banks, and Control Bank A this range is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 200 and 228 steps withdrawn inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these banks positions in these ranges satisfies all accident analysis assumptions concerning their position. The range for control Bank B is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 160 and 228 steps withdrawn inclusive. For Control Banks C and D the range is defined as the group demand counter indicated position between 0 and 228 steps withdrawn. Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits. The full out position will be specifically established for each cycle by the Reload Safety Analysis for that cycle. This position will be within the band established by "FULLY WITHDRAWN" and will be administratively controlled. This band is allowable to minimize RCCA wear, pursuant to Information Notice 87-19.

The ACTION statements which permit limited variation from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Mis-alignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a mis-aligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumption used in the accident analysis.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} >541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
License No. DPR-75

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

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated January 3, 1989 and supplemented by letter dated February 16, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 66 , 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 startup from the fifth refueling outage, currently scheduled to begin January 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 22, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 66

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-3	1-3
1-6	1-6
B 2-1	B 2-1
3/4 1-17	3/4 1-17
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
3/4 1-21	3/4 1-21
3/4 10-1	3/4 10-1
3/4 10-6	-
B 3/4 1-4	B 3/4 1-4
B 3/4 10-1	B 3/4 10-1

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."

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1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

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FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 228 steps withdrawn, inclusive. FULLY WITHDRAWN will be established by the current reload analysis.

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1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

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SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee, as shown in Figure 5.1-3, and which defines the exclusion area as shown in Figure 5.1-1.

SOLIDIFICATION

1.30 SOLIDIFICATION shall be the conversion of wet radioactive wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.31 SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.32 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,

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$ of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3(1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods FULLY WITHDRAWN to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power

POSITION INDICATION SYSTEM SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.2.2 The group demand position indicator shall be OPERABLE for each shutdown and control rod not fully inserted. During the performance of individual full length (shutdown and control) rod testing measurement during rod position indication system calibration:

a. Only one shutdown or control bank shall be withdrawn from the fully inserted position at a time, and

b. K_{eff} shall be maintained less than or equal to 0.95.

APPLICABILITY: MODES 3*, 4*, and 5*

ACTION:

With less than the above required group demand position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.2.2 Each of the above required group demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days.

*With the reactor trip system breakers in the closed position

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 228 steps withdrawn shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 & 2

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 76% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

POSITION INDICATION SYSTEM SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

APPLICABILITY: MODES 1*, and 2*#

ACTION:

With a maximum of one shutdown rod not FULLY WITHDRAWN, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. FULLY WITHDRAW the rod, or,

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators within one hour after rod motion:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3

#With K_{eff} greater than or equal to 1.0

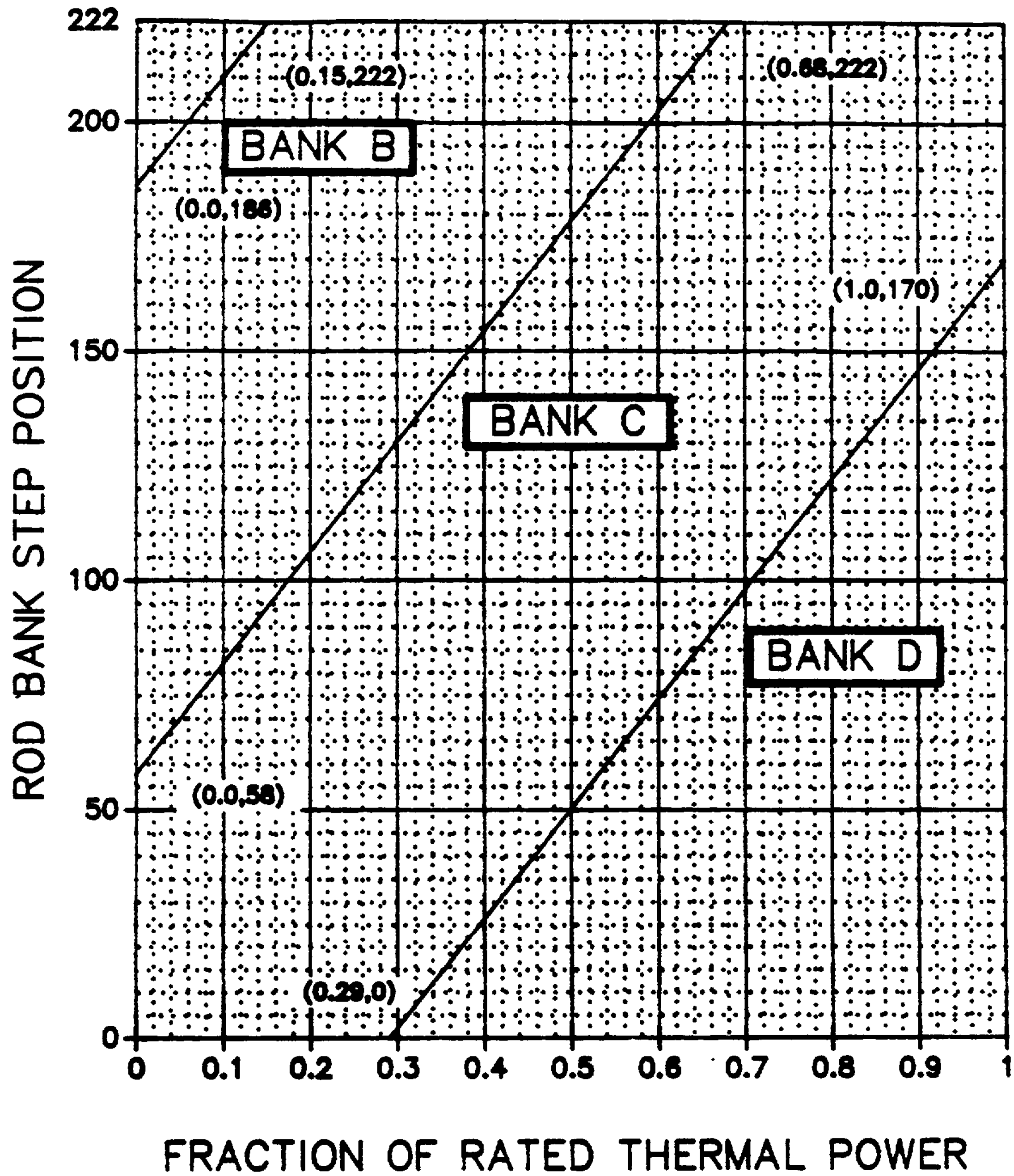

Figure 3.1-1
ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or FULLY WITHDRAWN shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod mis-alignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within ± 12 steps of the bank demand position for a range of positions. For the Shutdown Banks, and Control Bank A this range is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 200 and 228 steps withdrawn inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these banks positions in these ranges satisfies all accident analysis assumptions concerning their position. The range for control Bank B is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 160 and 228 steps withdrawn inclusive. For Control Banks C and D the range is defined as the group demand counter indicated position between 0 and 228 steps withdrawn. Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits. The full out position will be specifically established for each cycle by the Reload Safety Analysis for that cycle. This position will be within the band established by "FULLY WITHDRAWN" and will be administratively controlled. This band is allowable to minimize RCCA wear, pursuant to Information Notice 87-19.

The ACTION statements which permit limited variation from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Mis-alignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a mis-aligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumption used in the accident analysis.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} >541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth, and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the Reactor Coolant System T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the reactor core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is, at times, necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not be allowed by Specification 3.1.3.6 which may, in turn, cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 NO FLOW TESTS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 91 AND 66 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated January 3, 1989 and supplemented by letter dated February 16, 1989, Public Service Electric & Gas Company requested an amendment to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Generating Station, Unit Nos. 1 and 2. The supplemental letter provided additional clarifying information but did not change the original application. The proposed amendments would redefine the FULLY WITHDRAWN position of the rod control cluster assemblies (RCCAs) from 228 steps withdrawn to a band between 222 steps and 228 steps withdrawn. Other proposed changes are: delete Figure 3.1.2 from the Unit 1 Technical Specifications (TSs); delete Specification 3.10.5 from the Unit 2 TSs; and incorporate rod drop testing requirements into Specification 3.1.3.2.2 for Units 1 and 2.

2.0 EVALUATION

The licensee proposed the change to minimize localized rod cluster control assembly (RCCA) wear at the tip of the control rods. The proposed technical specification change will allow operation with the RCCAs inserted up to six steps (Step 222) from their normal withdrawn position (Step 228). The licensee has committed to defining the specific step between 222 and 228 as FULLY WITHDRAWN in the Reload Safety Analysis for that cycle. At 222 steps withdrawn the RCCAs are 3.5 inches into the active fuel region. Because the top region of the core has such low worth, the resultant power distribution perturbations are calculated to be negligible and can be accommodated with available margin. Similarly, the effect on shutdown margin is minimal (0.06% $\Delta\rho$) and can be accommodated by available excess shutdown margin (2.55% $\Delta\rho$, Unit 1 end of cycle 8; and 1.68% $\Delta\rho$, Unit 2 end of cycle 5). These calculations include the rods at step 228 plus other appropriate uncertainties and allowance for the most reactive rod stuck out. The Technical Specifications for shutdown margin is 1.6% $\Delta\rho$.

The impact on other key parameters was found to be negligible in the licensee's analysis. Because the proposed change will insert RCCAs so little into the active fuel region, the staff would expect essentially negligible effects of the proposed change as reported in the licensee's evaluation. As such the staff finds this proposed change to be acceptable.

Figure 3.1.2 in the Salem 1 Technical Specifications (TSs) provides the Rod Bank Insertion Limits versus Thermal Power for three-loop operation. Operation of Salem 1 with less than 4-loops has not been approved. Therefore, the licensee has proposed to delete Figure 3.1.2 from the Salem 1 TSs rather than modifying the curve to reflect the redefinition of FULLY WITHDRAWN. The staff finds this proposed change to be acceptable.

Special Test Exception (STE) 3.10.5 is an exception to Specification 3.1.3.2.2 that allows the Analog Rod Position Indication (ARPI) system to be inoperable during control rod drop time measurements. The bases of STE 3.10.5 states "The exception is required since the data necessary to determine rod drop times is derived from the induced voltage in position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the position indication systems remain OPERABLE". Amendment 73 to Salem 1 TSs and 48 to Salem 2 TSs changed Specification 3.1.3.2.2 to require the group demand position indicators to be OPERABLE in Modes 3, 4 and 5, thus eliminating the requirement to have the ARPI system OPERABLE in Modes 3, 4 and 5.

For this reason, STE 3.10.5 is no longer necessary. However, STE 3.10.5 did limit withdrawal of only one shutdown bank or one control bank at a time from the fully inserted position. Also, K_{eff} is to be maintained less than or equal to 0.95. In order to maintain these restrictions on control rod drop time tests, the licensee has proposed to add these restrictions to Specification 3.1.3.2.2 in the Salem 1 and 2 TSs. The staff finds this proposed change to be acceptable.

In addition to the above, typographical errors were corrected in the changed pages.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 6208) on February 8, 1989 and consulted with the State of New Jersey. No public comments were received and the State of New Jersey did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: J. Stone

Dated: March 22, 1989