February 6, 1997

Mr. Leon R. Eliason
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 (TAC NOS. M95509 AND M95510)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 191 and 174 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 31, 1996, as supplemented December 23, 1996.

These amendments (1) revise the reactor vessel level indication system action statements, (2) revise the channel calibration definition, and (3) delete a requirement to install a jumper in the auxiliary feedwater actuation logic.

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

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

Docket Nos. 50-272/311

Enclosures:
1. Amendment No. 191 to License No. DPR-70
2. Amendment No. 174 to License No. DPR-75
3. Safety Evaluation

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

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

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Units 1 and 2

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. DPR-70

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

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated May 31, 1996, as supplemented December 23, 1996, 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:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 191 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 6, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-1	1-1
3/4 3-22	3/4 3-22
3/4 3-55	3/4 3-55
3/4 3-56a	3/4 3-56a
6-24	6-24

1.0 DEFINITIONS

DEFINED TERMS

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

ACTION

1.2 ACTION shall be that part of a specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.3 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever an RTD or thermocouple sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

1.5 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

1.6 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.

TABLE 3.3-3 (Continued)

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

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. The Minimum Channels OPERABLE requirements is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.1.

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 6 hours or, be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may proceed provided that the inoperable channel is restored to OPERABLE within 72 hours.

ACTION 22 - NOT USED

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

TABLE 3.3-11 (CONTINUED)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
13. PORV Block Valve Position Indicator	2/valve**	1	1, 2
14. Pressurizer Safety Valve Position Indicator	2/valve**	1	1, 2
15. Containment Pressure - Narrow Range	2	1	1, 2
16. Containment Pressure - Wide Range	2	1	7, 2
17. Containment Water Level - Wide Range	2	1	7, 2
18. Core Exit Thermocouples	4/core quadrant	2/core quadrant	1, 2
19. Reactor Vessel Level Instrumentation System (RVLIS)	2	1	8, 9

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

TABLE 3.3-11 (continued)

TABLE NOTATION

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

ACTION 7 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed until the next CHANNEL CALIBRATION (which shall be performed upon the next entry into MODE 5, COLD SHUTDOWN).

ACTION 8 With one RVLIS channel inoperable, restore the RVLIS channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 9 With both RVLIS channels inoperable, restore one channel to OPERABLE status within 7 days or submit a special report in accordance with Specification 6.9.4.

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list of descriptions of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.9 The F_{xy} limits for Rated Thermal Power (F^{RTP}_{xy}) for all core planes containing bank "D" control rods and all unrodded core planes, and the plot of predicted heat flux hot channel factor times relative power ($F^{T}_{Q}*P_{REL}$) vs. Axial Core Height with the limit envelope shall be provided to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector. The Report shall be provided to the Commission upon issuance.

In addition, in the event that the limit should change, requiring a new submittal or submittal of an amended Peaking Factor Limit Report, it will be submitted upon issuance.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.

6.9.4. When a report is required by ACTION 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 174
License No. DPR-75

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

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated May 31, 1996, as supplemented December 23, 1996, 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. 174 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 6, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 174

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-1	1-1
3/4 3-23	3/4 3-23
3/4 3-51a	3/4 3-51a
3/4 3-51c	3/4 3-51c
6-24	6-24

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TABLE 3.3-11 (continued)

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e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

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6.9.4. When a report is required by ACTION 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 191 AND 174 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated May 31, 1996, as supplemented December 23, 1996, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit Nos. 1 and 2, Technical Specifications (TSs). The requested changes would (1) revise the reactor vessel level indication system (RVLIS), (2) revise the channel calibration definition to better address temperature detector channel calibration methodology, and (3) delete a requirement to install a jumper in the auxiliary feedwater actuation logic. The December 23, 1996, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Proposed TS Change To Revise RVLIS Action Statements

The licensee proposes to revise the action statements associated with RVLIS. RVLIS is part of the safety related display information that the operator uses to perform manual functions and to determine the effect of the manual actions following a reactor trip and subsequent shutdown. RVLIS monitors the reactor vessel during abnormal plant conditions by using differential pressure (d/p) transmitters to measure vessel level or relative void content of the fluid surrounding the core.

RVLIS was upgraded during the Unit 1 and Unit 2 refueling outages in 1992 and 1991. During the timeframe that RVLIS was being upgraded, a temporary variance at each unit allowed both RVLIS channels to be inoperable as long as the required channels for reactor coolant system subcooling margin monitor and the core exit thermocouples (T/C) were operable. This temporary variance, as

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discussed in detail in Amendment 117 for Unit 1 dated February 12, 1991, and Amendment 95 for Unit 2 dated September 10, 1990, has now expired on both units.

As indicated in the enclosures to the February 12, 1991 staff correspondence, the note at the bottom of Table 3.3-11 page 3/4 3-55, "Accident Monitoring Instrumentation," Amendment 117 for Unit 1 indicates that Action 8 remains in effect until startup from the 10th refueling outage at which time RVLIS will be upgraded, and upon expiration, Action Statements 1 and 2 will apply. As indicated in the enclosures to the September 10, 1990 staff correspondence, the note at the bottom of Table 3.3-11, page 3/4 3-51, "Accident Monitoring Instrumentation," Amendment 95 for Unit 2, indicates that Action 8 will remain in effect until startup from the 6th refueling outage at which time RVLIS will be upgraded, and upon expiration, Action Statements 1 and 2 will apply. Action Statement 1 indicates that with one channel of RVLIS accident monitoring instrumentation inoperable, restore the inoperable channel to operable status within 7 days or be in hot shutdown within the next 12 hours. Action statement 2 indicates that with two channels of RVLIS accident monitoring instrumentation inoperable, restore the inoperable channel(s) to operable status within 48 hours or be in hot shutdown within the next 12 hours.

The licensee proposes to change the present RVLIS TS action statements from a TS requirement which requires shutdown in 7 days if one channel is inoperable and shutdown in 48 hours if two channels are inoperable to a TS requirement which requires the promulgation of a special report if one channel is inoperable for 30 days and a special report if two channels are inoperable for 7 days. Thus, in the event of RVLIS inoperability, unit shutdown is not required. This proposed action is consistent with the guidelines of NUREG 1431, Revision 1, Westinghouse Standard Technical Specifications. The special report is appropriate in lieu of shutdown since (1) alternative actions are identified in the TS before the loss of RVLIS functional capability, and (2) there is a low probability of unit conditions that would require information provided by the RVLIS instrumentation in that its function is only required during reactor shutdown conditions.

The requirements for the special report will be delineated in a new Salem Units 1 and 2 administrative Technical Specification Special Reports Section 6.9.4. As indicated, the requirements for the special report in the administrative TSs are consistent with the action statements and reporting requirements for RVLIS as indicated in NUREG-1431, Revision 1, however, whereas the Westinghouse Standard TS states, "The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule ...," the licensee's proposed change states, "The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and

the plans and schedule ..." The licensee indicates that the phrase "for inadequate core cooling" was added to clarify the monitoring function required since the proposed change only pertains to RVLIS whose sole function is to monitor for inadequate core cooling.

The staff agrees with the licensee's clarification and finds the above discussed TS change to revise the RVLIS action statements to be acceptable.

2.2 Proposed TS Change To Revise The Channel Calibration Definition To Better Address Temperature Detector Channel Calibration Methodology

The licensee requested that Salem Units 1 and 2 TS Definition 1.4 pertaining to Channel Calibration be revised to replace the existing definition. The licensee proposes this revision to better account for temperature detector channel calibration methodology. The licensee indicated that the proposed change ensures the required testing methodology aligns with standard industry methodology for instrument channels having a T/C or resistance temperature detector (RTD) as a sensor in order to prevent unnecessary removal of these sensors. Removal and reinstallation of RTDs solely for the purpose of calibration could introduce errors and increase personnel exposure. The licensee currently performs a cross calibration of the reactor coolant RTDs on a refueling outage frequency.

The licensee's current practice for completing channel calibrations for instrument channels having RTD or T/C sensors is to perform an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel, which is consistent with standard industry practice.

The issue of cross calibration was addressed in NUREG/CR-5560, "Aging of Nuclear Plant Resistance Temperature Detectors," which recognizes that on-line cross calibration can be a reasonable method for RTD calibration. However, as stated in NUREG/CR-5560, to perform in-situ calibration would normally require one or more newly calibrated RTDs to be used as a reference. Without a reference, the cross calibration will not account for common mode (systematic) drift and will only provide information on the consistency and not the accuracy of the installed RTDs. The cross calibration technique assumes that the average of the RTD measurements represents the true process temperature and that RTD drift is random and not systematic. The results of studies referenced in NUREG/CR-5560 indicate that RTD drift is usually random. However, the particular testing done to validate the cross calibration methodology in NUREG/CR-5560 utilized newly calibrated RTDs for the test.

The licensee's proposed TS change is consistent with the manner in which NUREG-1431, Westinghouse STS, defines channel calibration. At present, the licensee performs a cross calibration every refueling outage, but without a freshly calibrated RTD or T/C. The licensee proposes that whenever an RTD or T/C sensing element is replaced, the next required channel calibration will

include an inplace cross calibration which compares the other sensing elements with the recently installed sensing element. This cross calibration methodology provides for an on-line calibration testing capability.

The staff agrees with the licensee's proposed change to the definition of channel calibration and finds it acceptable.

2.3 Proposed TS Change To Delete A Requirement To Install A Jumper In The Auxiliary Feedwater Actuation Logic

The licensee proposes to delete a TS requirement to install a jumper in the auxiliary feedwater actuation logic during certain required surveillance testing. This proposed TS change has been requested due to a design change in which the jumper function is now performed by an installed relay.

Previously, a requirement to install a jumper was added to Action Statement 21.b of Table 3.3-3 via Amendment 39 for Unit 1 and Amendment 116 for Unit 2. The requirement states, "If the affected main feedwater pump is expected to be out of service for more than 72 hours, the inoperable channel is jumpered so as to enable the start circuit of the auxiliary feedwater pump upon loss of the other main feedwater pump." This requirement ensured the actuation of auxiliary feedwater upon loss of the feedwater pumps. The licensee is implementing a permanent design modification to ensure auxiliary feedwater pump actuation, therefore, a jumper would no longer be required. The requirement for the jumper, delineated in Action Statement 21.b of Table 3.3-3 would be deleted.

The modification, an installed relay actuation, performs the same function as that of the manual jumper. The installation of the relay enables the auxiliary feedwater pump auto-start circuitry and precludes the need for manual installation of a jumper. Upon a steam generator feedwater pump trip, failsafe (de-energize to actuate) relays automatically perform the same function as the manual installation of the jumpers. The relay actuation condition is displayed to the operators by illumination of a control console pushbutton light for each steam generator feedwater pump, "SGFP TRIP-AFP AUTO ARMED."

The staff agrees with licensee's proposed TS change to delete Action Statement 21.b of Table 3.3-3 which requires the installation of a jumper based on the addition of a relay, and finds it acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. The State official, by letter dated June 18, 1996, had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC 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 (61 FR 30641). The amendments also relate to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). 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.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: February 6, 1997