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

January 5, 1993

Docket Nos. 50-272
and 50-311

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: REMOVAL OF FIRE PROTECTION PROGRAM ELEMENTS FROM TECHNICAL SPECIFICATIONS, SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M81236 AND M81237)

The Commission has issued the enclosed Amendment Nos. 139 and 117 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Units 1 and 2. These amendments consist of changes to licenses DPR-70 and DPR-75 and to the Technical Specifications (TSs) in response to your application dated August 2, 1991, and supplemented by letter dated September 30, 1992.

These amendments revise the fire protection license condition and relocate the fire protection technical specifications to plant procedures and to the Updated Final Safety Analysis Report in accordance with the guidance provided in Generic Letters 86-10 and 88-12. The Technical Evaluation Report includes a discussion of the inclusion of alternate shutdown components in the technical specifications as identified in Generic Letter 81-12. This discussion concludes that not all alternate shutdown components are currently included within the technical specifications. This issue will be addressed by separate correspondence between the NRC and the licensee.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. You are requested to notify the NRC, in writing, when these amendments have been implemented at Salem, Units 1 and 2.

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. 139 to License No. DPR-70
2. Amendment No. 117 to License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	MO'Brien(2)	CGrimes, 11E21	JWhite, RGN-I
NRC & Local PDRs	JStone	CMcCracken, 8D1	
PDI-2 Reading	OGC	ACRS(10)	
SVarga	DHagan, 3206	OPA	
JCalvo	GHill(8), P1-22	OC/LFMB	
CMiller	Wanda Jones, P-370	EWenzinger, RGN-I	

OFC	:PDI-2/LA MOB	:PDI-2/PM	:OGC	:PDI-2/D	:	:
NAME	:MO'Brien	:JStone:rb	: S. Hom	:CMiller	:	:
DATE	:12/21/92	:12/21/92	:12/29/92	:1/5/93	:	:

110061

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See next page

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

Salem Nuclear Generating Station,
Units 1 and 2

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Baltimore, MD 21202-3486

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
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 August 2, 1991, and supplemented by letter dated September 30, 1992, 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. 139 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Also, the license is amended by replacing Paragraph 2.C.(5) on page 4 of Facility Operating License DPR-70 with the following:*

(5) PSE&G shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:

PSE&G may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

4. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Page 4 of License DPR-70
2. Changes to the Technical Specifications

Date of Issuance: January 5, 1993

*Page 4 is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT 1 TO LICENSE AMENDMENT NO. 139

FACILITY OPERATING LICENSE NO. DRP-70

DOCKET NO. 50-272

Revise License DPR-70 as follows:

Remove Page	Insert Page
4	4

(1) Maximum Power Level

Public Service Electric and Gas Company is authorized to operate the facility at a steady state reactor core power level not in excess of 3338 megawatts (one hundred percent of rated core power). Prior to attaining the one hundred percent power level, Public Service Electric and Gas Company shall complete the preoperational tests, startup tests and other items identified in Attachment 1 to this amended license in the sequence specified. Attachment 1 is an integral part of this amended license.

(2) Technical Specifications

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

(3) Deleted Per Amendment 22, 11-20-79

(4) Less than Four Loop Operation

Public Service Electric and Gas Company shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of Appendix A to this license) with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensees and approval for less than four loop operation at power levels above P-7 has been granted by the Commission by Amendment of this license.

(5) PSE&G shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:

PSE&G may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 139

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
Index IV	Index IV
Index VII	Index VII
Index XIV	Index XIV
3/4 3-49	3/4 3-49 through 3/4 3-52
3/4 3-50	-
3/4 3-51	-
3/4 3-52	-
3/4 7-32	-
3/4 7-33	-
3/4 7-34	-
3/4 7-35	-
3/4 7-36	-
3/4 7-37	-
3/4 7-38	-
3/4 7-39	-
3/4 7-40	-
3/4 7-41	-
3/4 7-42	-
B 3/4 3-3	B 3-4 3-3
B 3/4 7-7	B 3-4 7-7
B 3/4 7-8	-

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 139

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
6-2	6-2
6-7	6-7
6-24	6-24
6-25	6-25
6-26	6-26
6-27	6-27
6-28	6-28
6-29	6-29
-	6-30

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

3/4.3.3.6 THIS SECTION DELETED

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

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

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

PAGES 3/4 3-49 THROUGH 3/4 3-52 ARE INTENTIONALLY BLANK

SNUBBERS (Continued)

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18-month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance program.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

FACILITY STAFF (Continued)

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, at least one licensed Senior Reactor Operator shall be in the Control Room area at all times.

c. All CORE ALTERATIONS shall be observed and directly supervised by a licensed Senior Reactor Operator who has no other concurrent responsibilities during this operation.

d. The amount of overtime worked by plant staff members performing safety-related functions must be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

e. The Operating Engineer, Senior Nuclear Shift Supervisors, and Nuclear Shift Supervisors shall each hold a senior reactor operator license. The Nuclear Control Operators shall hold reactor operator licenses.

f. The Operations Manager shall meet one of the following:

(1) Hold an SRO license.

(2) Have held an SRO license for a similar unit (PWR).

6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall serve in an advisory capacity to the Senior Nuclear Shift Supervisor on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.2.3.2 The Shift Technical Advisor shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971* for comparable positions except for the individual designated as the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the licensed operators who shall comply with the requirements of 10CFR55.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall:

1) be coordinated by each functional level manager (Department Head) at the facility and maintained under the direction of the Manager - Nuclear Training

2) meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 for all affected positions except licensed operators and

3) comply with the requirements of 10CFR55 for licensed operators.

* The Operations Manager shall meet one of the following:

1) Hold an SRO license.

2) Have held an SRO license for a similar unit (PWR).

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_{xy}^{RTP}) 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_Q^T*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.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

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

a. Records and logs of unit 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 EVENTS submitted to the Commission.

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

e. DELETED

f. Records of changes made to Operating Procedures required by Specification 6.8.1.

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.

j. Records of reviews performed for changes made to procedures or reviews of tests and experiments, pursuant to 10CFR50.59.

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

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report, pursuant to 10CFR50.59.

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

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. DELETED

k. Records of SORC meetings and activities of OSR (and meetings of its predecessor, the Nuclear Review Board).

l. Records for Environmental Qualification which are covered under the provisions of Paragraph 6.16.

m. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

n. Records of secondary water sampling and water quality.

o. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Exposure Permit**. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels* such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Senior Nuclear Shift Supervisor on duty and/or Senior Supervisor - Radiation Protection. Doors shall remain locked except during periods of access by personnel under an approved Radiation Work Permit which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels

* This is normally defined as the level at a distance of 18 inches from the source or accessible surface.

** Radiation Protection Personnel or personnel escorted by Radiation Protection Personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonable constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

    b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

    c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

    b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determination; and

    c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee initiated major changes to the radioactive waste system (liquid, gaseous and solid):

1. Shall be reported to the Commission in the UFSAR for the period in which the evaluation was reviewed by (SORC). The discussion of each change shall contain:

    a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10CFR50.59;

    b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the (SORC).

2. Shall become effective upon review and acceptance by the SORC.

6.16 ENVIRONMENTAL QUALIFICATION

6.16.1 All safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-70 dated October 24, 1980.

6.16.2 Complete and auditable records shall be available and maintained at a central location which describe the environmental qualification method used for all safety related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

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ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117
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 August 2, 1991, and supplemented by letter dated September 30, 1992, 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. 117 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Also, the license is amended by replacing Paragraph 2.C.(10) on page 7 with the following and adjusting pages 8-9-10 and 11 of Facility Operating License DPR-75:*

(10) Fire Protection

PSE&G shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:

> PSE&G may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

4. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
1. Pages 7, 8-9-10, and 11 of License DPR-75
2. Changes to the Technical Specifications

Date of Issuance: January 5, 1993

*Pages 7, 8-9-10, and 11 are attached, for convenience, for the composite license to reflect this change.

ATTACHMENT 1 TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise License DPR-75 as follows:

Remove Pages	Insert Pages
Page 7	Page 7
Page 8	-
Page 9	Page 8-9-10
Page 10	-
Page 11	Page 11

(d) Complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document complete compliance by June 30, 1982.

(e) Within 90 days of receipt of the equipment qualification safety evaluation, the licensee shall either (i) provide missing documentation identified in Sections 3 and 4 of the equipment qualification safety evaluation which will demonstrate compliance of the applicable equipment with NUREG-0588, or (ii) commit to corrective actions which will result in documentation of compliance of applicable equipment with NUREG-0588 no later than June 30, 1982.

(10) Fire Protection

PSE&G shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:

> PSE&G may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

PAGES 8, 9, AND 10 ARE INTENTIONALLY BLANK

(11) Containment Isolation (Section 6.2.3, Supplements 4 and 5)

Within 90 days after issuance of the license, PSE&G shall demonstrate to the satisfaction of the NRC that the present containment isolation provisions for the main feedwater lines comply with the requirements of General Design Criterion 57 under all postulated accident conditions, or propose a design change that will achieve compliance. If necessary, the design change shall be implemented during the first refueling outage.

(12) Main Condenser (Section 14.0, Supplement 4)

Prior to exceeding 50 percent power, PSE&G shall complete the preoperational testing of the remaining three of six circulators to be tested in the main condenser for the circulating water system.

(13) River Traffic Accidents (Section 2.2.1, Supplement 1)

PSE&G shall also report for the Salem facility any information reported for the Hope Creek facility relating to circumstances which suggest that the risk from flammable gas clouds (resulting from river traffic accidents on the Delaware River) varies significantly from that previously considered.

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
Index IV	Index IV
Index VII	Index VII
Index XIV	Index XIV
3/4 3-46	3/4 3-46 through 3/4 3-49
3/4 3-47	-
3/4 3-48	-
3/4 3-49	-
3/4 7-27 thru 3/4 7-30	-
3/4 7-31	-
3/4 7-32	-
3/4 7-33	-
3/4 7-34	-
3/4 7-35	-
3/4 7-36	-
3/4 7-37	-
3/4 7-38	-
3/4 7-39	-
B 3/4 3-2	B 3/4 3-2
B 3/4 7-7	B 3/4 7-7
B 3/4 7-8	-

ATTACHMENT 2 TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
6-2	6-2
6-7	6-7
6-24	6-24
6-25	6-25
6-26	6-26
6-27	6-27
6-28	6-28
6-29	6-29
-	6-30

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

PAGES 3/4 3-46 THROUGH 3/4 3-49 ARE INTENTIONALLY BLANK

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and normalizing its respective output.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$, a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

NOT REQUIRED

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

NOT REQUIRED

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

3/4.3.3.6

THIS SECTION DELETED

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the Recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

SNUBBERS (Continued)

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18-month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance program.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

FACILITY STAFF (Continued)

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, at least one licensed Senior Reactor Operator shall be in the Control Room area at all times.

c. All CORE ALTERATIONS shall be observed and directly supervised by a licensed Senior Reactor Operator who has no other concurrent responsibilities during this operation.

d. The amount of overtime worked by plant staff members performing safety-related functions must be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

e. The Operating Engineer, Senior Nuclear Shift Supervisors, and Nuclear Shift Supervisors shall each hold a senior reactor operator license. The Nuclear Control Operators shall hold reactor operator licenses.

f. The Operations Manager shall meet one of the following:

(1) Hold an SRO license.

(2) Have held an SRO license for a similar unit (PWR).

6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall serve in an advisory capacity to the Senior Nuclear Shift Supervisor on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.2.3.2 The Shift Technical Advisor shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971* for comparable positions except for the individual designated as the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the licensed operators who shall comply with the requirements of 10CFR55.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall:

1) be coordinated by each functional level manager (Department Head) at the facility and maintained under the direction of the Manager - Nuclear Training

2) meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 for all affected positions except licensed operators and

3) comply with the requirements of 10CFR55 for licensed operators.

* The Operations Manager shall meet one of the following:

1) Hold an SRO license.

2) Have held an SRO license for a similar unit (PWR).

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_{xy}^{RTP}) 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 ($F_Q^T * P_{REL}$) 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.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

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

a. Records and logs of unit 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 EVENTS submitted to the Commission.

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

e. DELETED

f. Records of changes made to Operating Procedures required by Specification 6.8.1.

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.

j. Records of reviews performed for changes made to procedures or reviews of tests and experiments, pursuant to 10CFR50.59.

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

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report, pursuant to 10CFR50.59.

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

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those facility components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. DELETED

k. Records of SORC meetings and activities of OSR (and meetings of its predecessor, the Nuclear Review Board).

l. Records for Environmental Qualification which are covered under the provisions of Paragraphs 2.C(7) and 2.C(8) of Facility Operating License DPR-75.

m. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

n. Records of secondary water sampling and water quality.

o. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Exposure Permit**. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Work Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels* such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Senior Nuclear Shift Supervisor on duty and/or Senior Supervisor - Radiation Protection. Doors shall remain locked except during periods of access by personnel under an approved Radiation Work Permit which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels

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* This is normally defined as the level at a distance of 18 inches from the source or accessible surface.

** Radiation Protection Personnel or personnel escorted by Radiation Protection Personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonable constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

    b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determination; and

    c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.15.1 Licensee initiated major changes to the radioactive waste system (liquid, gaseous and solid):

1. Shall be reported to the Commission in the UFSAR for the period in which the evaluation was reviewed by (SORC). The discussion of each change shall contain:

    a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10CFR50.59;

    b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the (SORC).

2. Shall become effective upon review and acceptance by the SORC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 139 AND 117 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 August 2, 1991, as supplemented by letter dated September 30, 1992, the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) submitted a request for changes to the Salem Nuclear Generating Station, Units 1 and 2, licenses and Technical Specifications (TS). The requested changes would revise the fire protection license condition and relocate the fire protection technical specifications to plant procedures and to the Updated Final Safety Analysis Report in accordance with guidance provided in Generic Letters 86-10 and 88-12. The September 30, 1992, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

By letters dated August 2, 1991 and September 30, 1992, the licensee requested an amendment to their operating license to revise the fire protection license condition and relocate fire protection technical specifications to plant procedures and to the Updated Final Safety Analysis Report in accordance with the guidance provided in Generic Letters 86-10 and 88-12. The staff's contractor, Science Applications International Corporation (SAIC), reviewed the licensee's submittal and found it acceptable with the exception of Technical Specification (TS) for alternate shutdown equipment. This issue will be pursued with the licensee independent of this request.

Attachment 1 to this safety evaluation (SE) is the Technical Evaluation Report (TER) from SAIC. The staff has reviewed the TER and agrees with the SAIC conclusions. The licensee's request for an amendment to their operating license is therefore granted. This TER constitutes the staff's SE for the relocation of the fire protection technical specifications and the change to the operating license condition.

Also, the licensee's submittal did not include a proposed revision to Page IV of the index for the Salem 1 TS. With the concurrence of the licensee, this change has been included. This is an administrative change and is 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 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 (57 FR 11116). 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.

This amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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.

Principal Contributor: A. Singh

Attachment: Technical Evaluation Report

Date: January 5, 1993

ATTACHMENT

Technical Evaluation Report
Fire Protection License Amendment Request
Salem Generating Station Unit Nos. 1 and 2
TAC Nos. M81236 and M81237

SAIC-92/6905

Science Applications International Corporation
An Employee-Owned Company

October 15, 1992

Prepared for:

U.S. Nuclear Regulatory Commission
Washington, DC 20555

Contract NRC-03-87-029
Task Order 143B

9210220182

1710 Goodridge Drive, P.O. Box 1303, McLean, Virginia 22102 (703) 821-4300
Other SAIC Offices: Albuquerque, Boston, Colorado Springs, Dayton, Huntsville, Las Vegas, Los Angeles, Oak Ridge, Orlando, Palo Alto, San Diego, Seattle, and Tucson

XA

Technical Evaluation Report
Fire Protection License Amendment Request
Salem Generating Station Unit Nos. 1 and 2
TAC Nos. M81236 and M81237

SAIC-92/6905

October 15, 1992

Prepared for:

U.S. Nuclear Regulatory Commission
Washington, DC 20555

Contract NRC-03-87-029
Task Order 143B

1.0 INTRODUCTION

This Technical Evaluation Report (TER) documents an independent review of a request for amendment of Facility Operating License DPR-70 and DPR-75 for Salem Generating Station (SGS), Unit Nos. 1 and 2. This request was submitted to the Nuclear Regulatory Commission (NRC) by Public Service Electric and Gas, the licensee, by letter dated August 2, 1991. A meeting was held at the site on June 30, 1992 to discuss issues raised in the preliminary review. As a result of this meeting, a number of items were identified which required additional information or clarification from the licensee. The licensee provided this information in a submittal to the NRC dated September 30, 1992.

In addition, a review of existing technical specifications was made to determine if specific guidance contained in NRC Generic Letter 81-12 had been followed. Specifically, the presence of surveillance and operability requirements for alternate shutdown equipment was reviewed.

2.0 DISCUSSION

The licensee has requested an amendment to the Unit 1 and Unit 2 Operating Licenses which would revise the fire protection license condition and relocate the surveillance and operability requirements relating to fire protection currently contained in technical specifications, to plant periodic test procedures. The basis for implementation of the program will be defined in the Updated Facility Safety Analysis Report (UFSAR). This amendment is being requested by the licensee in accordance with guidance provided in NRC Generic Letter 86-10.

The current technical specifications for SGS do not address many plant modifications that have taken place. In addition, the licensee has requested that certain requirements with current technical specifications be modified to be more consistent with Westinghouse Standard Technical specifications and/or the more recently NRC approved fire protection program for the adjacent Hope Creek Generating Station. In order to address these issues and to respond to NRC Guidance for removal of fire protection technical specifications as delineated in Generic Letter 86-10, the licensee has provided a two-step submittal. The licensee submittal, which was prepared in a format as recommended by the NRC Project Manager (PM), first provides a modification to existing SGS technical specifications to bring them up to date and address consistency issues. Then, the licensee requests that these updated technical specifications be removed and equivalent requirements be placed in plant operating procedures. The end result is that the following changes to the Facility Operating License and newly revised technical specifications have been requested:

Add the following as replacements for the entire current license Condition 2.C.(5) to the Salem Unit 1 Facility Operating License and License condition 2.C.(10) of the Salem Unit No. 2 Facility Operating License:

> PSE&G shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis report, and as approved in the NRC Safety Evaluation Report dated November 20, 1979, and in its supplements, subject to the following provision:
>
> > PSE&G may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

From both Salem Units' technical specifications, remove in their entirety, Sections 3/4.3.3.6 (Fire Detection Instrumentation), Table 3.3-10 (Fire Detection Instruments), 3/4.7.10 (Fire Suppression Systems), 3/4.7.11 (Penetration Fire Barriers), Specification 6.2.2.e and the reference to Fire Brigade in associated Footnote "#" (Facility Staff), Specification 6.4.2 (Training), Bases sections B3/4.3.3.6, B3/4.7.10, and B3/4.7.11, and index page references to the foregoing sections.

Additionally, insert the following Special Report requirements as a new technical specification Section 6.9.3:

> "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, Docket Control Desk, Washington, DC 20555, with a copy to the Regional Administrators the Regional office of the NRC via the License Event report System within 30 days."

3.0 EVALUATION

The two separate portions of the licensee's submittal were reviewed as part of this Technical Evaluation Report. First, the proposed modifications to existing technical specifications were reviewed to ensure that any modifications are acceptable and secondly, that the requested deletion of the revised technical specifications is in accordance with guidance provided in Generic Letters 86-10 and 88-12. In addition, an assessment was made against the guidance in Generic Letter 81-12 which establishes the need to provide technical specification requirements for alternate shutdown equipment which was not previously contained in plant technical specifications,

Although Generic Letter 88-12 states that licensees should not use this FSAR incorporation as an opportunity to make changes in the approved Fire Protection Program, the licensee has proposed changes to the existing technical specifications in addition to the removal request. However, this has been done based on discussions between the licensee and the NRC Project Manager in order to correct and update a number of inconsistencies with the Salem Technical Specifications. The licensee had made a previous submittal to update certain fire protection sections of the Salem technical specifications, however, the change request was not approved by the NRC pending a complete updating of the fire protection portion of the technical specifications. Therefore, the August 2, 1991 submittal as supplemented by the September 30, 1992 submittal is an attempt by the licensee to completely update the Salem technical specifications prior to complying with the Guidance in Generic Letters 86-10 and 88-12. Due to the need to make a significant number of changes to the Salem requirements, this approach is considered to be the most appropriate.

The August 2, 1991 submittal, as modified by the September 30, 1992 submittal, contain a significant number of proposed changes to the Unit 1 and Unit 2 technical specifications. The primary reasons for the proposed changes are to incorporate new fire protection systems, make the Salem fire protection surveillance requirements consistent with Revision 4 of the Westinghouse Standard Technical Specifications and to be as consistent as possible with the approved requirements at Hope Creek Generating Station. Since the Salem and Hope Creek plants are directly adjacent to each other and are serviced by a single fire protection organization, the goal of consistency is considered prudent and an acceptable bases for change. In addition, the attempt to bring the Salem Requirements in line with the Standard Technical Specifications is considered appropriate. Several proposed changes vary from standard technical specifications, however are consistent with the previously approved Hope Creek program and do not represent a reduction in the level of fire protection safety at Salem. However, one proposed change was identified which did not correspond to standard technical specifications or the approved Hope Creek program. This change permits the use of chains and locks to secure fire protection valves inside containment in lieu of physically verifying the valve position. The licensee sites ALARA concerns in addition to overriding personnel safety issues. Since these valves are in a normally inaccessible area and physical controls in the form of locks and chains are provided, subjecting personnel to the additional concerns associated with a containment entry would not significantly increase the level of control over the valves and therefore is not considered necessary to provide an adequate level of fire protection.

A detailed description of all the proposed technical specification changes was provided in the August 2, 1991 submittal. Based on discussions with plant personnel, NRC requested additional clarification of certain changes. The licensee provided these additional clarifications in the September 30, 1992 submittal. Based on a review of this information, including justification for the valve surveillance issue discussed above, the proposed technical specifications are considered to be an enhancement of the existing fire protection program and are therefore acceptable.

A review of the proposed technical specifications was also made to ensure that the need to review changes to the fire protection program by the plant operating review committee was maintained. It was determined that this requirement is included in the appropriate sections. In addition, the need to report violations to the fire protection program which would have adversely affected the ability to achieve safe shutdown has been added. This is consistent with the guidance contained in Generic Letters 86-10 and 88-12.

Attachment 4 to the August 2, 1991 submittal contains proposed changes to the fire protection program description contained in the updated Final Safety Analysis Report (UFSAR). These changes include establishing the UFSAR as the basis of the Salem fire protection program. Reference to plant surveillance and operability procedures rather than technical specifications is also included. After a review of these proposed UFSAR changes, it is determined that the intent of Generic Letters 86-10 and 88-12 is satisfied.

A review of plant surveillance and operability procedures intended to take the place of technical specifications was performed during the site visit. Although, many of the procedures have not completed the formal review process, it was apparent that, upon completion and approval, the surveillance and operability procedures are intended to maintain the same level of safety as established by the plant technical specifications.

A discussion was held with licensee representatives during the site visit concerning the inclusion of alternate shutdown components within technical specifications as identified in Generic Letter 81-12. Based on this discussion and as confirmed in the September 30, 1992 response, not all alternate shutdown components are included within technical specifications. Although, the licensee did identify that plant procedures are provided to ensure operability of alternate shutdown components not included in technical specifications. Since the licensee has not included technical specifications for all alternate shutdown components, they have not complied with the guidance provided in NRC Generic Letter 81-12.

4.0 CONCLUSION

Based on the evaluation of the licensee's submittal dated August 2, 1991 and as clarified in the September 30, 1992 response, the proposed modifications to the fire protection portions for Salem Unit Nos. 1 and 2 are considered acceptable. In addition, the removal of these updated technical specifications by placing the equivalent requirements in plant procedures, and by referencing these procedures in the UFSAR, is in accordance with Generic Letters 86-10 and 88-12. Therefore, the request to amend Facility Operating Licenses DPR-70 and DPR-75 by adding the standard license condition provided in Generic Letter 86-10 and removing the appropriate fire protection sections from the technical specifications can be granted. The technical specifications which are to be used as a basis for the new plant surveillance and operability requirements are the revised technical specifications provided in the August 2, 1991 submittal as revised by the September 30, 1992 submittal.

Based on a review of the plant technical specifications against the guidance in Generic letter 81-12, it is concluded that the licensee has not provide technical specifications for all alternate shutdown components and has therefore not met the guidance provided in Generic Letter 81-12.