

November 10, 1975

Dockets Nos. 50-277
and 50-278

Philadelphia Electric Company
ATTN: Mr. Edward G. Bauer, Jr., Esquire
Vice President and General Counsel
2301 Market Street
Philadelphia, Pennsylvania 19101

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 12 and 10 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units 2 and 3. These amendments include Changes Nos. 13 and 10 to the Technical Specifications and are in response to your request dated December 16, 1974.

The amendments incorporate into the Peach Bottom Atomic Power Station, Units 2 and 3 Technical Specifications changes to the Administrative Controls. Changes to your proposal were necessary to meet our requirements. These have been discussed with your staff. The technical specifications are based on the regulatory positions described in Guides 1.8, "Personnel Selection and Training", 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4, and 1.33, "Quality Assurance Program Requirements".

We request that you use the formats presented in the Appendices to Regulatory Guide 1.16, Revision 4, for reporting operating information and that you report events of the type described under the section "Events of Potential Public Interest". Instructions for using these reporting formats are contained in Regulatory Guide 1.16 (a copy is enclosed for your use), and AEC report OOE-SS-001 titled "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File" (a copy of which was provided you previously). This report is modified by updated instructions dated August 21, 1975 which are enclosed. Copy requirements are summarized in Regulatory Guide 10.1, "Compilation of Reporting Requirements for Persons Subject to NRC Regulations", a copy of which is also enclosed. This guide will assist you in identifying reports that are required by the Commission's regulations set forth in Title 10 Code of Federal Regulations but are not contained in your Technical Specifications. Reports that are required by the regulations have not been repeated in your Technical Specifications.

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Dockets Nos. 50-277
and 50-278

Chase R. Stephens
Docketing and Service Section
Office of the Secretary of the Commission

FEDERAL REGISTER NOTICE

Enclosed for your transmission to the Office of the Federal Register for filing and publication are two signed originals of a Federal Register Notice as follows:

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

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

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY
OPERATING LICENSES

Twelve copies of the above notice also are enclosed for your use and distribution to the Public Document Room.

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

Enclosure:
As Stated

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SURNAME	CParrish:acr	GLear				
DATE	11/ /75	11/ /75				

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

Sincerely,

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

Enclosures:

1. Amendment No. 12
2. Amendment No. 10
3. Regulatory Guide 1.16
4. Updated Instructions
5. Regulatory Guide 10.1
6. Safety Evaluation
7. Federal Register Notice

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

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

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. DPR-56

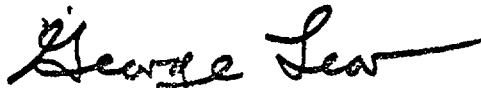
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company (the licensees), dated December 16, 1974, complies 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; and
 - 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.
2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.(c).2 of Facility License No. DPR-56 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 10."

3. This license amendment is effective ¹⁻⁴⁻⁷⁶ 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:
Change No. 10 to the
Technical Specifications

Date of Issuance: November 10, 1975

*See Appendix B
11/12/75*

ATTACHMENT TO LICENSE AMENDMENT NO. 10
CHANGE NO. 10 TO THE TECHNICAL SPECIFICATIONS
FACILITY OPERATING LICENSE NO. DPR-56
DOCKET NO. 50-278

Replace pages iii, iv, v, and vi with the attached revised pages (no change on page v).

Replace pages 1 and 2 with the attached revised pages 1 and 2 to Appendix A.

Replace Section 6.0 Administrative Controls in its entirety and replace with the revised section.

Replace page 48 and page 51 with the attached revised pages to Appendix B.

PBAPS

TABLE OF CONTENTS (Cont'd)

Page No.

<u>LIMITING CONDITIONS FOR OPERATION</u>	241
MAJOR DESIGN FEATURES	243
ADMINISTRATIVE CONTROLS	243
6.1 Responsibility	243
6.2 Organization	243
6.3 Facility Staff Qualifications	246
6.4 Training	246
6.5 Review and Audit	251
6.6 Reportable Occurrence Action	251
6.7 Safety Limit Violation	251
6.8 Procedures	252
6.9 Reporting Requirements	260
6.10 Record Retention	261
6.11 Radiation Protection Program	261
6.12 Respiratory Protection Program	263
6.13 High Radiation Area	

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
1.1.1	Limiting Power Distribution	24
2.1.1	APRM Flow Biased Scram and Rod Blocks	25
2.1.2	Total Peaking Factor	26
2.1.3	APRM Flow Bias Scram Vs. Reactor Core Flow	27
4.1.1	Instrument Test Interval Determination Curves	55
4.2.2	Probability of System Unavailability Vs. Test Interval	98
3.4.1	Required Volume and Concentration of Standby Liquid Control System Solution	122
3.4.2	Required Temperature Vs. Concentration for Standby Liquid Control System Solution	123
3.6.1	RPV Pressurization Temperature Limits Vs. Neutron Exposure	164
6.2-1	Management Organization Chart	244
6.2-2	Organization for Conduct of Plant Operations	245

PBAPS

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
3.1.1	Reactor Protection System (Scram) Instrumentation Requirement	37
4.1.1	Reactor Protection System (Scram) Instrument Functional Tests	41
4.1.2	Reactor Protection System (Scram) Instrument Calibration	44
3.2.A	Instrumentation That Initiates Primary Containment Isolation	61
3.2.B	Instrumentation That Initiates or Controls the Core and Containment Cooling Systems	64
3.2.C	Instrumentation That Initiates Control Rod Blocks	73
3.2.D	Radiation Monitoring Systems That Initiate and/or Isolate Systems	75
3.2.E	Instrumentation That Monitors Drywell Leak Detection	76
3.2.F	Surveillance Instrumentation	77
3.2.G	Instrumentation That Initiates Recirculation Pump Trip	79
4.2.A	Minimum Test and Calibration Frequency for PCIS	80
4.2.B	Minimum Test and Calibration Frequency for CSCS	81
4.2.C	Minimum Test and Calibration Frequency for Control Rod Blocks Actuation	83
4.2.D	Minimum Test and Calibration Frequency for Radiation Monitoring Systems	84
4.2.E	Minimum Test and Calibration Frequency for Drywell Leak Detection	85
4.2.F	Minimum Test and Calibration Frequency for Surveillance Instrumentation	86
4.2.G	Minimum Test and Calibration Frequency for Recirculation Pump Trip	88
4.6.1	In-Service Inspection Program	150

PBAPS

LIST OF TABLES (Cont'd.)

<u>Table</u>	<u>Title</u>	<u>Page</u>
3.7.1	Primary Containment Isolation Valves	179
3.7.2	Testable Penetrations With Double O-Ring Seals	184
3.7.3	Testable Penetrations With Testable Bellows	184
3.7.4	Primary Containment Testable Isolation Valves	185
4.8.1	Radioactive Liquid Waste Sampling and Analysis	210
4.8.2	Radioactive Gaseous Waste Sampling and Analysis	211
6.12-1	Protection Factors for Respirators	265

PBAPS

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

- A. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary systems are assured. Exceeding such a limit requires unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.
- B. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.
- C. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.
- D. Deleted

1.0 DEFINITIONS (Cont'd)

10

DELETED

- E. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.
- F. Operating - Operating means that a system or component is performing its intended functions in its required manner.
- G. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.
- H. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup" or "Run" position with the reactor critical and above 1% rated power.
- I. Hot Standby Condition - Hot standby condition means operation with coolant temperature greater than 212°F, system pressure less than 1055 psig, and the mode switch in the Startup/Hot Standby position. The main steam isolation valves may be opened to provide steam to the reactor feed pumps.
- J. Cold Condition - Reactor coolant temperature equal to or less than 212°F.
- K. Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.
- L. Cold Shutdown - The reactor is in the shutdown mode, the reactor coolant temperature equal to or less than 212°F, and the reactor vessel is vented to atmosphere.
- M. Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the

6.0 ADMINISTRATIVE CONTROLS

6.1 Responsibility

6.1.1 The Station Superintendent shall be responsible for overall facility operation. In the absence of the Station Superintendent, the Assistant Superintendent or the Engineer-Technical (or any other person that the Station Superintendent may designate in writing) shall, in that order, assume the Superintendent's responsibility for overall facility operation.

6.2 Organization

Offsite

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

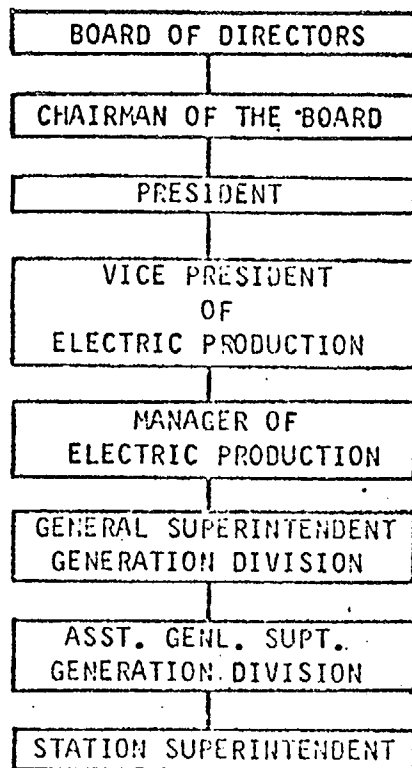
Facility Staff

6.2.2 The facility organization shall be as shown on Figure 6.2.-2 and:

- a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Figure 6.2.-2.
- b. At least one licensed operator shall be in the control room and assigned to each reactor that contains fuel.
- c. At least two licensed operators, excluding the operator on the second unit, shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.
- d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.
- e. ALL CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

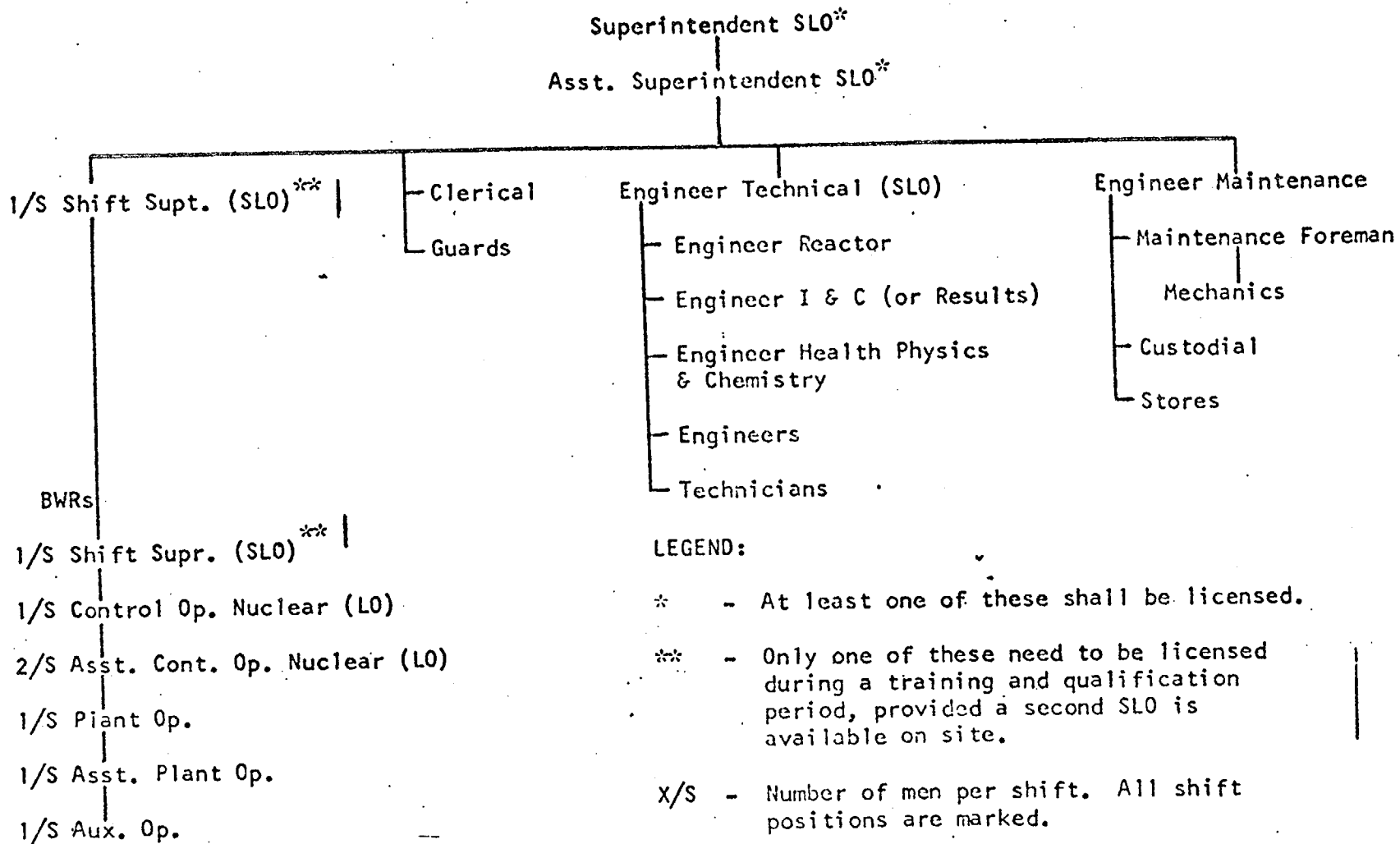
6.3 Facility Staff Qualifications

6.3.1 Each member of the facility staff shall meet the minimum qualifications of ANSI N18.1-1971 for comparable positions.



MANAGEMENT ORGANIZATION
CHART

FIGURE 6.2-1



LEGEND:

- * - At least one of these shall be licensed.
- ** - Only one of these need to be licensed during a training and qualification period, provided a second SLO is available on site.
- X/S - Number of men per shift. All shift positions are marked.
- LO - NRC Licensed Operator
- SLO - NRC Licensed Senior Operator

ORGANIZATION FOR CONDUCT OF
PLANT OPERATIONS

FIGURE 6.2-2

6.4 Training

- 6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Station Superintendent and shall meet the requirements of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55. 10

6.5 Review and Audit

6.5.1 Plant Operation Review Committee (PORC)

Function

- 6.5.1.1 The Plant Operation Review Committee shall function to advise the Station Superintendent on all matters related to nuclear safety.

Composition

- 6.5.1.2 The Plant Operation Review Committee shall be composed of the:

- Station Superintendent - Chairman
- Station Assistant Superintendent
- Engineer - Technical
- Engineer - Maintenance
- Engineer - Results
- Engineer - Reactor
- Engineer - Instrument & Control
- Engineer - Health Physics & Chemistry
- Shift Superintendent

Alternates

- 6.5.1.3 Alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in PORC activities at any one time.

Meeting Frequency

- 6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman.

Quorum

- 6.5.1.5 A quorum of the PORC shall consist of the Chairman and four members or their alternates. 10

Responsibilities

- 6.5.1.6 The Plant Operation Review Committee shall be responsible for:
- a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Station Superintendent to affect nuclear safety.

6.5.1.6 Continued

- b. Review of all proposed tests and experiments that affect nuclear safety.
- c. Review of all proposed changes to the Technical Specifications.
- d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.
- e. Investigation of all violations of the Technical Specifications and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the General or Assistant General Superintendent, Generation Division and the Chairman of the Operation and Safety Review Committee.
- f. Review of facility operations to detect potential safety hazards.
- g. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Operation and Safety Review Committee.
- h. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Plan, to the Chairman of the Operation and Safety Review Committee. | 10
- i. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Plan, to the Chairman of the Operation and Safety Review Committee. | 10

Authority

6.5.1.7 The Plant Operation Review Committee shall:

- a. Recommend to the Station Superintendent written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.
- b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6 (a) through (e) above constitutes an unreviewed safety question, as defined in Section 50.59, 10 CFR. | 10
- c. Provide immediate written notification to the General or Assistant General Superintendent, Generation Division and the Operation and Safety Review Committee of disagreement between the PORC and the Station Superintendent; however, the Station Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

Records

- 6.5.1.8 The Plant Operation Review Committee shall maintain written minutes of each meeting and copies shall be provided to the General or Assistant General Superintendent, Generation Division and Chairman of the

6.5.1.8 Continued

Operation and Safety Review Committee.

6.5.2 Operation and Safety Review Committee

Function

6.5.2.1 The Operation and Safety Review Committee shall function to provide independent review and audit of designated activities in the area of:

- a. nuclear power plant operations.
- b. nuclear engineering
- c. chemistry and radiochemistry
- d. metallurgy
- e. instrumentation and control
- f. radiological safety
- g. mechanical and electrical engineering
- h. quality assurance practices

(the members of the OSR Committee will be competent in the area of quality assurance practice and cognizant of the Quality Assurance requirements of 10 CFR 50 Appendix B. Additionally, they will be cognizant of the corporate Quality Assurance Program and will have the corporate Quality Assurance Organization available to them.)

Composition

6.5.2.2 The Operation and Safety Review Committee shall be composed of the:

- Manager - Electric Production Department - Chairman
- General Superintendent - Maintenance Division
- Director - Environmental & Laboratory Division
- Manager - Engineering & Research Department
- Chief Mechanical Engineer
- Chief Electrical Engineer
- Assistant Director - Research Division

Alternates

6.5.2.3 Alternate Members shall be appointed in writing by the OSR Committee Chairman. Each permanent member shall have a designated alternate to serve in his absence, and a current list of these alternates shall be maintained in Board records. Each alternate member will serve on a continuing basis.

Consultants

- 6.5.2.4 Consultants shall be utilized as determined by the OSR Committee Chairman to provide expert advice to the OSR Committee.

Meeting Frequency

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

Quorum

- 6.5.2.6 A quorum of the OSR Committee shall consist of the Chairman or his designated alternate and four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

Review

- 6.5.2.7 The OSR Committee shall review:
- a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.
 - b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.
 - c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.
 - d. Proposed changes in Technical Specifications or licenses.
 - e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.
 - f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.
 - g. All events which are required by regulations or Technical Specifications to be reported to the NRC in writing within 24 hours.
 - h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.
 - i. Reports and meeting minutes of the Plant Operation Review Committee.

Audits

- 6.5.2.8 Audits of facility activities shall be performed under the cognizance of the OSR Committee. These audits shall encompass:
- a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.
 - b. The performance, training and qualifications of the entire facility staff at least once per year.
 - c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.
 - d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two years.
 - e. The Facility Emergency Plan and implementing procedures at least once per two years.
 - f. The Facility Security Plan and implementing procedures at least once per two years.
 - g. Any other area of facility operation considered appropriate by the OSR Committee or the Vice President, Electric Production.

Authority

- 6.5.2.9 The OSR Committee shall report to and advise the Vice President, Electric Production on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

Records

- 6.5.2.10 Records of OSR Committee activities shall be prepared, approved and distributed as indicated below:
- a. Minutes of each OSR Committee meeting shall be prepared, approved and forwarded to the Vice President, Electric Production within 14 days following each meeting.
 - b. Reports of reviews encompassed by Section 6.5.2.7.e, f, g and h above, shall be prepared, approved and forwarded to the Vice President, Electric Production within 14 days following completion of the review.
 - c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Vice President, Electric Production and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 Reportable Occurrence Action

6.6.1 The following actions shall be taken in the event of a Reportable Occurrence:

- a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.
- b. Each Reportable Occurrence Report submitted to the Commission shall be reviewed by the PORC and submitted to the OSR Committee and the General or Assistant General Superintendent, Generation Division.

6.7 Safety Limit Violation

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

- a. The provisions of 10 CFR 50.36 (c) (1) (i) shall be complied with immediately.
- b. The Safety Limit violation shall be reported to the Commission, the General or Assistant General Superintendent, Generation Division and to the OSR Committee immediately.
- c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.
- d. The Safety Limit Violation Report shall be submitted to the Commission, the OSR Committee and the General or Assistant General Superintendent, Generation Division within 10 days of the violation.

6.8 Procedures

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet the requirements of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Station Superintendent prior to implementation and periodically as set forth in each document.

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

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

6.8.3 Continued

- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.
- c. The change is documented, reviewed by the PORC and approved by the Station Superintendent within 7 days of implementation.

6.9 Reporting Requirements

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

6.9.1 Routine Reports

- a. Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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

- b. Annual Operating Report^{1/}. Routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience gained during the year, even though some repetition of previously reported

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

6.9.1 Continued

information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

- (1) A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintenance not covered in item 1.b.(2)(e) below.
- (2) For each outage or forced reduction in power^{2/} of over twenty percent of design power level where the reduction extends for greater than four hours:
 - (a) the proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);
 - (b) a brief discussion of (or reference to reports of) any reportable occurrences pertaining to the outage or power reduction;
 - (c) corrective action taken to reduce the probability of recurrence, if appropriate;
 - (d) operating time lost as a result of the outage or power reduction (for scheduled or forced outages,^{3/} use the generator off-line hours; for forced reductions in power, use the approximate duration of operation at reduced power);
 - (e) a description of major safety-related corrective maintenance performed during the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction; and
 - (f) a report of any single release of radioactivity or radiation exposure specifically associated with the outage which accounts for more than 10% of the allowable annual values.

^{2/}The term "forced reduction in power" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance and calibration activities requiring power reductions are not covered by this section.

^{3/}The term "forced outage" is normally defined in the electric power industry as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

6.9.1 Continued

- (3) A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,^{4/} e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.
- (4) Indications of failed fuel resulting from irradiated fuel examinations, including eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the appropriate Regional Office, to arrive no later than the tenth of each month following the calendar month covered by the report.

6.9.2 Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Region Office, or his designate no later than the first working day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

^{4/}This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

6.9.2 Continued

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

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 2.a(5), 2.a(6), or 2.b(1) below.

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

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the technical specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 2.b(2) below.

- (3) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

- (4) Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions greater than or equal to 1.0% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

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

- (6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

Note: For items 2.a(5) and 2.a(6) reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 2.b(2) and 2.b(3) below.

6.9.2 Continued

- (7) Conditions arising from natural or man-made events that, as a direct result of the event require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.
- (8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.
- (9) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

b. Thirty Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

- (1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.
- (2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in items 2.b(1) and 2.b(2) need not be reported except where test results themselves reveal a degraded mode as described above.

6.9. Continued

- (3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.
- (4) Abnormal degradation of systems other than those specified in item 2.a(3) above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in technical specifications need not be reported under this item.

6.9.4 Unique Reporting Requirements ,

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

- a. Loss of shutdown margin, Specification 3.3.A and 4.3.A.
- b. Reactor vessel inservice inspection, Specification 3.6.G and 4.6.G.
- c. Secondary Containment leak rate testing, Specification 4.7.C upon completion of the test.
- d. Primary Containment leak rate testing, Specification 4.7.A upon completion of the test.
- e. Release rate of Radioactive Effluents, Specification 3.8.B.7, 3.8.C.3.b, 3.8.C.5.
- f. Effluent Releases

Effluent data should be summarized monthly, except in instances when more detailed data is needed, and the items listed below reported semi-annually on the standard form "Report of Radioactive Effluents".

(1) Gaseous Releases

- (a) total radioactivity released (in curies) of noble and activation gases.
- (b) maximum noble gas release rate during any one-hour period.

6.9.3 Continued

(c) total radioactivity released (in curies) by nuclide, based on representative isotopic analyses performed.

(d) percent of technical specification limit.

(2) Iodine Releases

(a) total (I-131, I-133, I-135) radioactivity released (in curies).

(b) total radioactivity released (in curies) by nuclide, based on representative isotopic analyses performed.

(c) percent of technical specification limit.

(3) Particulate Releases

(a) gross alpha radioactivity released (in curies) excluding background radioactivity.

(b) total radioactivity released (in curies) of nuclides with half-lives greater than eight days.

(c) percent of technical specification limit.

(4) Liquid Releases

(a) gross radioactivity ($\beta\gamma$) released (in curies) and average concentration released to the unrestricted area.

(b) total tritium and alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

(c) total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

(d) total volume (in liters) of liquid waste released.

(e) total volume (in liters) of dilution water used during release from the restricted area.

(f) the maximum concentration of group radioactivity ($\beta - \gamma$) released to the unrestricted area (averaged over the period of release).

(g) total radioactivity (in curies) released by nuclide, based on representative isotopic analyses performed.

(h) percent of technical specification limit for total activity releases.

6.9.3 Continued

g. Solid Radioactive Waste

- (1) total volume (in cubic feet) of solid waste involved,
- (2) gross curie activity involved,
- (3) dates and disposition of the material, if shipped off-site.

h. Environmental Monitoring

(This portion of the annual operations report is exempted from the 60 day reporting period if required to allow environmental consultants sufficient time to generate their report).

- (1) Descriptive material covering the off-site environmental surveys performed during the reporting period including information on:

The number and types of samples taken; e.g., air, domestic water, marine life, milk, crops.

The number and types of measurements made; e.g., dosimetry.

Locations of the sample points and monitoring stations.

The frequency of the surveys.

A summary of survey results.

- (2) If a particular sample of measurement indicates statistically significant levels of radioactivity above established or concurrent backgrounds, the following information shall be provided:

The type of analysis performed; e.g., alpha, beta, gamma, and/or isotopic.

The minimum sensitivity of the monitoring system.

The measured radiation level of sample concentration.

The specific times when samples were taken and measurements were made.

An estimate of the likely resultant exposure to the public.

6.10 Record Retention**6.10.1** The following records shall be retained for at least five years:

- a. Records and logs of facility operation covering time interval at each power level.
- b. Records and logs of principle maintenance activities, inspections, repair and replacement of principle items of equipment related to nuclear safety.
- c. Reportable Occurrence Reports.
- d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
- e. Records of reactor tests and experiments.
- f. Records of changes made to Operating Procedures.
- g. Records of radioactive shipments.
- h. Records of sealed source leak tests and results.
- i. Records of annual physical inventory of all source material of record.

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

- a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.
- b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.
- c. Records of facility radiation and contamination surveys.

6.10.2 Continued

- 4. Records of radiation exposure for all individuals entering radiation control areas.
- 5. Records of gaseous and liquid radioactive material released to the environs.
- 6. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.
- 7. Records of training and qualification for current members of the plant staff.
- 8. Records of in-service inspections performed pursuant to these Technical Specifications.
- 9. Records of Quality Assurance activities required by the QA Manual, except as described in 6.10.1 above.
- 10. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.
- 11. Records of meetings of the PORC and the OSR Committee.

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 Respiratory Protection ProgramAllowance

6.12.1 Pursuant to 10 CFR 20.103 (c) (1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this facility in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

- 4. The limits provided in Section 20.103(a) and (b) shall not be exceeded.
- 5. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures

6.12.1 b. Continued

to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table I, Column 1, of 10 CFR 20.

- c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in §20.101. These materials shall be subject to applicable process and other engineering controls.

Protection Program

6.12.2 In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

- a. The limits specified in 6.12.1 above, are not exceeded.
- b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.12-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.
- c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.
- d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:
 1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper

6.12.2 d. Continued

- selection of respiratory protective equipment.
2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.
 3. Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.
 4. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.
 5. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.
 6. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.
- e. The licensee shall use equipment approved by the U.S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.12-1. Equipment not approved under U.S. Bureau of Mines^{5/} Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U.S. Bureau of Mines^{5/} approved equipment of the same type, as specified in Table 6.12-1.
- f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

Revocation

- 6.12.3 The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

6.13 High Radiation Area

- 6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c) (2) of 10 CFR 20:
- a. 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 Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

^{5/}and/or National Institute for Occupational Safety and Health (NIOSH).

6.13.1 Continued

- b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1 (a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of Shift Superintendent or Shift Supervisor.

TABLE 6.12-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES ¹	PROTECTION FACTORS ²	GUIDES TO SELECTION OF EQUIPMENT
		PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE ³	BUREAU OF MINES 5/ APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask ^{4,7}	NP	5	21B 30 CFR § 14.4(b)(4)
Facepiece, full ⁷	NP	100	21B 30 CFR § 14.4(b)(5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR § 12.2(c)(2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR § 12.2(c)(2) Type C(i)
Facepiece, full ⁷	D	100	19B 30 CFR § 12.2(c)(2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR § 12.2(c)(2) Type C(iii)
Hood	CF	5	6
Suit	CF	5	6
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full ⁷	D	100	13E 30 CFR § 11.4(b)(2)(i)
Facepiece, full	PD	1,000	13E 30 CFR § 11.4(b)(2)(ii)
Facepiece, full	R	100	13E 30 CFR § 11.4(b)(1)
III. COMBINATION RESPIRATOR			
Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19B CFR § 12.2(e) or applicable schedules as listed above

1, 2, 3, 4, 5, 6, and 7 (These notes are on the following pages)

5/and/or National Institute for Occupational Safety and Health (NIOSH).

TABLE 6.12-1 (Continued)

¹ See the following symbols:

- CF: continuous flow
- D: demand
- NP: negative pressure (i.e., negative phase during inhalation)
- PD: pressure demand (i.e., always positive pressure)
- R: recirculating (closed circuit)

² (a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

- (i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.
- (ii) for air-purifying respirators only when high efficiency (above 99.9% removal efficiency by U.S. Bureau of Mines type dioctyl phthalate (DOP) test) particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.
- (iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

³ Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air purifying respirators are not recommended for use against tritium oxide. See also footnote ⁵, below, concerning supplied-air suits and hoods.

⁴ Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR Part 20.

⁵ Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

PBAPS

TABLE 6.12-1 (Continued)

6 No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

7 Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines^{5/} according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U.S. Bureau of Mines^{5/} in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10 CFR 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

The program has not detected any changes in the radiological environment associated with the operation of Unit No. 1.

Samples are taken of the atmospheric, terrestrial and aquatic environment, using those media which are sensitive indicators of changes in the environment radioactivity such as particulate matter in air, water, soil and sediment, as well as those which could enter the human food chain, such as potable water, milk, vegetation and fish.

A program was initiated in 1970 to determine the concentration of tritium in Conowingo Pond in accordance with the requirement of Radwaste Permit No. 6769204 issued by the Commonwealth of Pennsylvania on January 6, 1970, and the results will become a part of the environmental monitoring data.

A program has also begun to monitor ambient gamma radiation using thermoluminescent dosimeters both on and around the Peach Bottom site. This program began in May 1971 with 8 stations, and is now being expanded through the addition of more monitoring locations to cover the region surrounding the site.

Through 1972, approximately 11,000 samples have been analyzed in the overall Peach Bottom Radiation Monitoring Program.

As per Section 7.1.2, Philadelphia Electric Company will periodically review the environmental radiation monitoring program. Results of this program will be reported as per Section 7.4.

7.0 Administrative Controls

7.1 Organization, Review and Audit

7.1.1 Organization

- A. The Plant Superintendent is responsible for the operation of the facility and to assure that the facility operates within the limits set forth in the environmental technical specifications.
- B. In all matters pertaining to operation of the facility and to the environmental technical specifications, the Plant Superintendent shall report to, and consult with, the General Superintendent - Generation Division. The management organization is shown in Figure 7.1-1.

7.1.2 Review and Audit

Committees for review and audit of plant operation are described in Section 6.5 of Appendix A to the Operating License.

7.4 Plant Reporting Requirements7.4.1 Routine Reports

In addition to the environmental monitoring information required by Section 6.9.3.h. of Appendix A to the Operating License, the following information shall be submitted in an annual report: | 10

- A. Records of special study programs data and analysis thereof.
- B. Records of changes to the plant which affect the environmental impact of the facility.
- C. Records of changes to environmental permits and certificates.

7.4.2 Non-Routine ReportsA. Environmental Deviation Reports

In the event of an environmental deviation as defined in the environmental technical specifications, notification shall be made within 24 hours by telephone or telegraph to the Director of the NRC Regional Inspection and Enforcement Office. A written report shall follow within 10 days to the Director, Office of Nuclear Reactor Regulation (copy to the Director of Regional Inspection and Enforcement Office).

The written report on an environmental deviation, and to the extent possible, the preliminary telephone and telegraph notification, should: (a) describe, analyze, and evaluate implications, (b) determine the cause of the occurrence and (c) indicate the corrective action (including any significant changes made in procedures) taken to preclude repetition of the occurrence and to prevent similar occurrences involving similar components or systems.

B. Reporting of Changes to the Plant or Permits

A written report, including an evaluation of the environmental impact resulting from a change, shall be forwarded to the Director, Office of Nuclear Reactor Regulation (copy to the Director of the Regional Inspection and Enforcement Office) in the event of:

1. Changes to the plant that affect the environmental impact evaluation contained in the Environmental Report or the Environmental Statement. This requirement does not preclude making changes on short notice that are minor in terms of environmental impact.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 12 TO FACILITY LICENSE NO. DPR-44 AND
AMENDMENT NO. 10 TO FACILITY LICENSE NO. DPR-56
CHANGES NOS. 13 AND 10 TO TECHNICAL SPECIFICATIONS

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

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

Introduction

By letter dated December 16, 1974, Philadelphia Electric Company proposed changes to the Technical Specifications appended to Facility Operating Licenses Nos. DPR-44 and DPR-56, for the Peach Bottom Atomic Power Station, Units 2 and 3. The proposed changes involve changes to the administrative controls including changes to the reporting requirements.

Discussion

The proposed changes would be administrative in nature and would affect the conduct of operation. The proposed changes are intended to provide uniform license requirements. Areas covered by the proposed uniform specifications include licensee staffing qualifications and management procedures involved with operating the reactor, reporting requirements, abnormal occurrence definition change, and a respiratory protection program.

Members of the facility staff should meet the requirements set forth in Guide 1.8, "Personnel Selection and Training" which endorses proposed ANSI N18.1, which was subsequently issued as ANSI N18.1-1971. Provisions for independent review of facility operations should be in accord with Guide 1.33, "Quality Assurance Program Requirements" which endorses proposed standard ANS 3.2, which was subsequently issued as ANSI 18.7-1972.

In Section 208 of the Energy Reorganization Act of 1974 "abnormal occurrences" is defined as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety. The term "abnormal occurrence" is reserved for usage by NRC. Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4, enumerates required reports consistent with Section 208. The proposed change to required reports identifies the reports required of all licensees not already identified by the regulations and those unique to this facility. The proposal would formalize present reporting and would delete any reports no longer needed for assessment of safety related activities. In addition, a radiation protection program delineates use of respiratory equipment in the event personnel are to be exposed to concentrations in excess of Part 20 concentrations.

Evaluation

The new guidance for reporting operating information does not identify any event as an "abnormal occurrence". The proposed reporting requirements also delete reporting of information no longer required and duplication of reported information. The standardization of required reports and desired format for the information will permit more rapid recognition of potential problems.

Identifying minimum acceptable qualifications for facility personnel should assure capable performance from the facility staff. Other administrative requirements also restated by the specifications assure uniformity and conformance to the desired features in the review, staffing, and procedures. Incorporating the currently accepted respiratory protection program at this time assures that a consistent method of using respiratory equipment is immediately available whenever needed. Similar changes are being approved for all power reactor licensees, so all licensees will have the same requirements presented in a uniform manner.

During our review of the proposed changes, we found that certain modifications to the proposal were necessary to have conformance with the desired regulatory position. These changes were discussed with your staff and have been incorporated into the proposal.

We have concluded that the proposal as modified improves the licensee's program for evaluating plant performance and the reporting of the operating information needed by the Commission to assess safety related activities and is acceptable. The facility staff qualifications and training program conform to Guide 1.8 and therefore are acceptable. The administrative procedures and facility review and audit are consistent

with Guide 1.33 and are acceptable. The modified reporting program is consistent with the guidance provided by Regulatory Guide 1.16, "Reporting of Operation Information - Appendix A Technical Specifications", Revision 4. The administrative controls are consistent with requirements being incorporated in Technical Specifications for new licensed facilities.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: November 10, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

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

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY
OPERATING LICENSES

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 12 and 10 to Facility Operating Licenses Nos. DPR-44 and DPR-56, respectively, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station, Units 2 and 3, located in Peach Bottom, York County, Pennsylvania. The amendment is effective 30 days from the date of issuance.

The amendment incorporates into the Peach Bottom Atomic Power Station, Units 2 and 3 Technical Specifications, changes to the Administrative Controls.

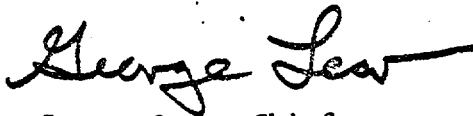
The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated December 16, 1974, (2) Amendments Nos. 12 and 10 to Licenses Nos. DPR-44 and DPR-56, respectively, with Changes Nos. 13 and 10, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Martin Memorial Library, 159 E. Market Street, York, Pennsylvania 17401.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 10th day of November, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION



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