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BScharf (10)
JMcGough
DEisenhut
ACRS (16)
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Dockets Nos. 50-277
and 50-278

DEC 1 8 1977

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. 27 and 27 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station, Units Nos. 2 and 3. These amendments consist of changes to the Technical Specifications and are in response to your requests dated August 26, 1976; May 6, 1977; September 21, 1977 and October 31, 1977. Your request to revise torus temperature logging requirements is still under review and is not included in these amendments.

These amendments will (1) delete the requirements for an Annual Operating Report in order to be consistent with Commission guidance; (2) revise the plant organization and the membership of the Plant Operation Review Committee (PORC) to reflect the addition of a Plant Engineer-Operations to the facility staff; and (3) revise other sections of the Technical Specifications where the changes are either editorial or pro forma in nature.

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

Original signed by
George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment Nos. 27 and 27
2. Safety Evaluation
3. FEDERAL REGISTER Notice

const 1

*SEE PREVIOUS YELLOW FOR CONCURRENCES

cc w/enclosures: See page 2

OFFICE➤			ORB #3	ORB #3	OELD	ORB #3
SURNAME➤			CParrish*	*DVerrelli:mjf	*	GLear
DATE➤			12/1/77	12/1/77	12/8/77	12/ /77

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624 cc:

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. DPR-44

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

   A. The applications 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 August 26, 1976; May 6, 1977; September 21, 1977; and October 31, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 37, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 13, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 37

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages 8 and 78 are also provided to maintain document completeness. No changes were made on page 78. Add pages 254a and 254b.

Remove	Replace
1 thru 8	1 thru 8
77	77
78	78
152	152
243	243
245 thru 254	245 thru 254
--	254a
--	254b

1.0 DEFINITIONS

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

Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud.

Normal control rod movement with the control drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a core alteration.

Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

Cold Condition - Reactor coolant temperature equal to or less than 212°F.

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.

Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958)

Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a

1.0 DEFINITIONS (Cont'd)

core spray pump to verify that it runs and that it pumps the required volume of water).

Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

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.

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.

Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

Limiting Conditions for Operations (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.

Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate

1.0 DEFINITIONS (Cont'd)

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.

Logic - A logic is an arrangement of relays, contacts and other components that produce a decision output.

(a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

(b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

Maximum Total Peaking Factor - The Maximum Total Peaking Factor (MTPF) is the lowest Total Peaking Factor which limits a fuel type to a Linear Heat Generation Rate (LHGR) corresponding to the operating limit at 100% power.

Minimum Critical Power Ratio (MCPR) - The minimum in-core critical power ratio corresponding to the most limiting fuel assembly in the core.

Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided: Refuel Mode, Run Mode, Shutdown Mode, Startup/Hot Standby Mode.

Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

1.0 DEFINITIONS (Cont'd)

Operating - Operating means that a system or component is performing its intended functions in its required manner.

Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, is 105% of rated power, which corresponds to 3440 MWt.

1.0 DEFINITIONS (Cont'd)

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.

Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

Refuel Mode - With the mode switch in the refuel position interlocks are established so that only one control rod may be withdrawn when the Source Range Monitors indicate at least 3 cps and the refueling crane is not over the reactor; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

Run Mode - In this mode the reactor system pressure is at or above 850 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

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 Nuclear Regulatory 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.

Secondary Containment Integrity - Secondary Containment integrity means that the reactor building is intact and the following conditions are met:

1.0 DEFINITIONS (Cont'd)

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All Reactor Building ventilation system automatic isolation valves are operable or deactivated in the isolation position.

Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig.

Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

Startup/Hot Standby Mode - In this mode the reactor protection scram trips, initiated by condenser low vacuum and main steam line isolation valve closure, are bypassed when reactor pressure is less than 1055 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the Startup/Hot Standby position of the mode switch.

Surveillance Frequency - Periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval. Surveillance tests are not required on systems or parts of systems that are not required to be operable or are tripped. If tests are missed on parts not required to be

1.0 DEFINITIONS (Cont'd)

operable or are tripped, then they shall be performed prior to returning the system to an operable status.

Total Peaking Factor - The ratio of the maximum fuel rod surface heat flux in an assembly to the average surface heat flux of the core.

Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation or protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

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TABLE 3.2.F

SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	Action
2	Reactor Water Level	Recorder 0-60" Indicator 0-60"	(1) (2) (3)
2	Reactor Pressure	Recorder 0-1500 psig Indicator 0-1200 psig	(1) (2) (3)
2	Drywell Pressure	Recorder 0-70 psig	(1) (2) (3)
2	Drywell Temperature	Recorder 0-400°F Indicator 0-400°F	(1) (2) (3)
2	Suppression Chamber Water Temperature	Recorder 0-600°F Indicator 0-400°F	(1) (2) (3)
2	Suppression Chamber Water Level	Recorder 0-25 ft. Indicator 0-25 ft.	(1) (2) (3)
1	Control Rod Position	28 Volt Indicating Lights	(1) (2) (3) (4)
1	Neutron Monitoring	SRM, IRM, LPRM 0-100%	(1) (2) (3) (4)

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

3.6.A & 4.6.A BASES (Cont'd.)

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutron ($\geq$1 mev) above about 10^{17} nvt may shift the NDT temperature of the vessel base metal above the initial value. Extensive tests have established the magnitude of changes as a function of the integrated neutron exposure. These changes presented in Figure 3.6.1 based on an initial maximum NDTT of the reactor vessel shell and head of 40°F. Test results as indicated in Appendix K of the FSAR show that the initial NDTT is less than this value.

Current NRC bases indicate that the vessel pressure should be limited when the vessel temperature is below 185°F. Other investigations indicate that this limit is conservative. This matter is currently under technical review by the applicable Code Committees. Based on this technical review, the applicant will submit a special report within five years which will provide the bases to review this limit as required.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be removed and tested to experimentally verify the values used for Figure 3.6.1.

As described in paragraph 4.2.5 of the Safety Analysis report, detailed stress analyses have been made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these transients are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50°F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

The plant safety analyses (Ref.: NEDO-21578) states that all MSIV valve closure - Flux scram is the event which satisfies the ASME Boiler and Pressure Code requirements for protection from the consequences of pressure in excess of the vessel design pressure. The reactor vessel pressure code limit of 1375 psig, given in Subsection 4.2 of the FSAR, is well above the peak pressure produced by the above overpressure event. Pressure transients and overpressurization events are analyzed assuming a maximum initial dome pressure of 1020 psig. A safety limit of 1020 psig will assure that the reactor operating pressure will not exceed the initial pressure assumed in the ASME vessel code compliance analysis.

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.

Amendment No. ~~12~~, 37

245

- Superintendent SLO*
  - Asst. Superintendent SLO*
    - Operations Engineer (SLO)
      - 1/S Shift Supt. (SLO)**
        - BWRs
      - 1/S Shift Supr. (SLO)**
      - 1/S Control Op. Nuclear (LO)
      - 2/S Asst. Cont. Op. Nuclear (LO)
      - 1/S Plant Op.
      - 1/S Asst. Plant Op.
      - 1/S Aux. Op.
    - Clerical
    - Guards
    - Engineer Technical (SLO)
      - Engineer Reactor
      - Engineer I & C (or Results
      - Engineer Health Physics & Chemistry
      - Engineers
      - Technicians
    - Engineer Maintenance
      - Maintenance Foreman
        - Mechanics
      - Custodial
      - Stores

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

6.3 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.

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 10 CFR 55, Appendix A.

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 - Operations
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.

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 Station Superintendent to affect nuclear safety.

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 Superintendent, Generation Division-Nuclear 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.

6.5.1.6 Continued

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.

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.

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 (c) above constitutes an unreviewed safety question, as defined in 10 CFR 50.59.

c. Provide immediate written notification to the Superintendent, Generation Division-Nuclear or, in his absence, the Superintendent, Generation Division--Fossil-Hydro, 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 Superintendent, Generation Division-Nuclear and Chairman of the Operation and Safety Review Committee.

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 Committee 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 advise 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 10 CFR 50.59, 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 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

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
Superintendent-Services Division
Manager-Engineering & Research Department
Chief Mechanical Engineer
Chief Electrical Engineer
Assistant Director-Research Division

6.5.2.7 Continued

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.

<u>Audits</u>

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 occuring 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 10 CFR 50, Appendix B, at least once per two years.

6.5.2.8 Continued

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 to 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 Superintendent, Generation Division-Nuclear.

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 Superintendent, Generation Division-Nuclear or, in his absence, the Superintendent, Generation Division--Fossil-Hydro 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 Superintendent, Generation Division-Nuclear within 14 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.

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 14 days of implementation.

6.9 <u>Reporting Requirements</u>

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 <u>Routine Reports</u>

a. <u>Startup Report</u>. 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 hyraulic 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 reported in this report.

6.9.1.a Continued

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 the 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 Occupational Exposure Tabulation[1]

A tabulation shall be made 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 function,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation shall be submitted for the previous calendar year prior to March 1 of each year. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling 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 job functions.

(1) A single submittal may be made for a multiple unit station.

(2) This tabulation supplements the requirements of 10 CFR 20.407.

6.9.1 Continued

c. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, and a narrative summary of the operating experience shall be submitted on a monthly basis to the Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the 15th of the 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 the occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the orignal 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 ten working days. The written followup report shall 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.

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 NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. DPR-56

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

   A. The applications 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 August 26, 1976; May 6, 1977; September 21, 1977; and October 31, 1977, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 37, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 13, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 37

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the are of chanqe. The corresponding overleaf pages 8 and 78 are also provided to maintain document completeness. No changes were made on page 78. Add pages 254a and 254b.

Remove	Replace
1 thru 8	1 thru 8
77	77
78	78
157	157
243	243
245 thru 254	245 thru 254
--	254a
--	254b

1.0 DEFINITIONS

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

Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud.

Normal control rod movement with the control drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation and the traversing in-core probe is not defined as a core alteration.

Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combined in logic.

Cold Condition - Reactor coolant temperature equal to or less than 212°F.

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.

Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the GEXL correlation. (Reference NEDO-10958)

Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required in response to accidents.

Functional Tests - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a

1.0 DEFINITIONS (Cont'd)

core spray pump to verify that it runs and that it pumps the required volume of water).

Hot Shutdown - The reactor is in the shutdown mode and the reactor coolant temperature greater than 212°F.

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.

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.

Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors.

Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

Limiting Conditions for Operations (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.

Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate

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1.0 DEFINITIONS (Cont'd)

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.

Logic - A logic is an arrangement of relays, contacts and other components that produce a decision output.

(a) Initiating - A logic that receives signals from channels and produces decision outputs to the actuation logic.

(b) Actuation - A logic that receives signals (either from initiation logic or channels) and produces decision outputs to accomplish a protective action.

Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit to insure all components are operable per design intent. Where practicable, action will go to completion; i.e., pumps will be started and valves operated.

Maximum Total Peaking Factor - The Maximum Total Peaking Factor (MTPF) is the lowest Total Peaking Factor which limits a fuel type to a Linear Heat Generation Rate (LHGR) corresponding to the operating limit at 100% power.

Minimum Critical Power Ratio (MCPR) - The minimum in-core critical power ratio corresponding to the most limiting fuel assembly in the core.

Mode of Operation - A reactor mode switch selects the proper interlocks for the operational status of the unit. The following are the modes and interlocks provided: Refuel Mode, Run Mode, Shutdown Mode, Startup/Hot Standby Mode.

Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

1.0 DEFINITIONS (Cont'd)

Operating - Operating means that a system or component is performing its intended functions in its required manner.

Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform necessary operational activities.
2. At least one door in each airlock is closed and sealed.
3. All automatic containment isolation valves are operable or deactivated in the isolated position.
4. All blind flanges and manways are closed.

Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

Rated Power - Rated power refers to operation at a reactor power of 3,293 MWt; this is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear system pressure refer to the values of these parameters when the reactor is at rated power. Design power, the power to which the safety analysis applies, is 105% of rated power, which corresponds to 3440 MWt.

1.0 DEFINITIONS (Cont'd)

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.

Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detectors.

Refuel Mode - With the mode switch in the refuel position interlocks are established so that only one control rod may be withdrawn when the Source Range Monitors indicate at least 3 cps and the refueling crane is not over the reactor; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig. If the refueling crane is over the reactor, all rods must be fully inserted and none can be withdrawn.

Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

Run Mode - In this mode the reactor system pressure is at or above 850 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

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 Nuclear Regulatory 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.

Secondary Containment Integrity - Secondary Containment integrity means that the reactor building is intact and the following conditions are met:

1.0 DEFINITIONS (Cont'd)

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All Reactor Building ventilation system automatic isolation valves are operable or deactivated in the isolation position.

Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

Shutdown Mode - Placing the mode switch to the shutdown position initiates a reactor scram and power to the control rod drives is removed. After a short time period (about 10 sec), the scram signal is removed allowing a scram reset and restoring the normal valve lineup in the control rod drive hydraulic system; also, the main steam line isolation scram and main condenser low vacuum scram are bypassed if reactor vessel pressure is below 1055 psig.

Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

Startup/Hot Standby Mode - In this mode the reactor protection scram trips, initiated by condenser low vacuum and main steam line isolation valve closure, are bypassed when reactor pressure is less than 1055 psig, the reactor protection system is energized with IRM neutron monitoring system trip, the APRM 15% high flux trip, and control rod withdrawal interlocks in service. This is often referred to as just Startup Mode. This is intended to imply the Startup/Hot Standby position of the mode switch.

Surveillance Frequency - Periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval. Surveillance tests are not required on systems or parts of systems that are not required to be operable or are tripped. If tests are missed on parts not required to be

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1.0 DEFINITIONS (Cont'd)

operable or are tripped, then they shall be performed prior to returning the system to an operable status.

Total Peaking Factor - The ratio of the maximum fuel rod surface heat flux in an assembly to the average surface heat flux of the core.

Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation or protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

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TABLE 3.2.F

SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	Action
2	Reactor Water Level	Recorder 0-60" Indicator 0-60"	(1) (2) (3)
2	Reactor Pressure	Recorder 0-1500 psig Indicator 0-1200 psig	(1) (2) (3)
2	Drywell Pressure	Recorder 0-70 psig	(1) (2) (3)
2	Drywell Temperature	Recorder 0-400°F Indicator 0-400°F	(1) (2) (3)
2	Suppression Chamber Water Temperature	Recorder 0-600°F Indicator 0-400°F	(1) (2) (3)
2	Suppression Chamber Water Level	Recorder 0-25 ft. Indicator 0-25 ft.	(1) (2) (3)
1	Control Rod Position	28 Volt Indicating Lights	(1) (2) (3) (4)
1	Neutron Monitoring	SRM, IRM, LPRM 0-100%	

NOTES FOR TABLE 3.2.F

(1) From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

(2) From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

(3) If the requirements of notes (1) and (2) cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Condition within 24 hours.

(4) These surveillance instruments are considered to be redundant to each other.

3.6.D & 4.6.D BASES

Safety and Relief Valves

The safety and relief valves are required to be operable above the pressure (122 psig) at which the core spray system is not designed to deliver full flow. The pressure relief system for each unit at the Peach Bottom APS has been sized to meet two design bases. First, the total safety/relief valve capacity has been established to meet the overpressure protection criteria of the ASME code. Second, the distribution of this required capacity between safety valves and relief valves has been set to meet design basis 4.4.4.1 of subsection 4.4 which states that the nuclear system design valves shall prevent opening of the safety valves during normal plant isolations and load rejections.

The details of the analysis which shows compliance with the ASME code requirements is presented in subsection 4.4 of the FSAR and the Reactor Vessel Overpressure Protection Summary Technical Report presented in Appendix K of the FSAR.

Eleven safety/relief valves and two safety valves have been installed on Peach Bottom Unit No. 3 with a total capacity of 78.0% of rated steam flow. The analysis of the worst overpressure transient, (3 second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure of 1279 psig if a neutron flux scram is assumed. This results in a 96 psig margin to the code allowable overpressure limit of 1375 psig.

To meet the power generation design basis, the total safety/relief capacity of 78.1% has been divided into 64.5% relief (11 valves) and 13.6% safety (2 valves). The analysis of the plant isolation transient (Turbine trip with bypass valve failure to open) assuming a turbine trip scram is presented in NEDO-21363 for Peach Bottom Unit No. 3. This analysis shows that the 11 relief valves limit pressure at the safety valves to 27 psig below the setting of the safety valves. Therefore, the safety valves will not open.

Experience in relief and safety valve operation shows that testing of 50 per cent of the valves per year is adequate to detect failures from deterioration. The relief and safety valves are benchtested every second

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.

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- Superintendent SLO*
  - Asst. Superintendent SLO*
    - Operations Engineer (SLO)
      - 1/S Shift Supt. (SLO)**
      - BWRs
      - 1/S Shift Supr. (SLO)**
      - 1/S Control Op. Nuclear (LO)
      - 2/S Asst. Cont. Op. Nuclear (LO)
      - 1/S Plant Op.
      - 1/S Asst. Plant Op.
      - 1/S Aux. Op.
    - Clerical
    - Guards
    - Engineer Technical (SLO)
      - Engineer Reactor
      - Engineer I & C (or Results
      - Engineer Health Physics & Chemistry
      - Engineers
      - Technicians
    - Engineer Maintenance
      - Maintenance Foreman
        - Mechanics
      - Custodial
      - Stores

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

6.3 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.

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 10 CFR 55, Appendix A.

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 - Operations
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.

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 Station Superintendent to affect nuclear safety.

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 Superintendent, Generation Division-Nuclear 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.

6.5.1.6 Continued

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.

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.

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 (c) above constitutes an unreviewed safety question, as defined in 10 CFR 50.59.

c. Provide immediate written notification to the Superintendent, Generation Division-Nuclear or, in his absence, the Superintendent, Generation Division--Fossil-Hydro, 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 Superintendent, Generation Division-Nuclear and Chairman of the 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
Superintendent-Services 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 Committee 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 advise 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 10 CFR 50.59, 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 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

6.5.2.7 Continued

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.

<u>Audits</u>

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 occuring 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 10 CFR 50, Appendix B, at least once per two years.

6.5.2.8 Continued

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 to 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 Superintendent, Generation Division-Nuclear.

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 Superintendent, Generation Division-Nuclear or, in his absence, the Superintendent, Generation Division--Fossil-Hydro 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 Superintendent, Generation Division-Nuclear within 14 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.

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 14 days of implementation.

6.9 <u>Reporting Requirements</u>

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 <u>Routine Reports</u>

a. <u>Startup Report</u>. 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 hyraulic 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 reported in this report.

6.9.1.a Continued

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 the 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 Occupational Exposure Tabulation[1]

A tabulation shall be made 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 function,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. This tabulation shall be submitted for the previous calendar year prior to March 1 of each year. The dose assignment to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling 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 job functions.

(1) A single submittal may be made for a multiple unit station.

(2) This tabulation supplements the requirements of 10 CFR 20.407.

6.9.1 Continued

c. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, and a narrative summary of the operating experience shall be submitted on a monthly basis to the Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the appropriate Regional Office, to be submitted no later than the 15th of the 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 the occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the orignal 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 ten working days. The written followup report shall 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 37 AND 37 TO

FACILITY LICENSE NOS. DPR-44 AND DPR-56

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

Introduction

By letters dated August 26, 1976, May 6, 1977, September 21, 1977 and October 31, 1977, Philadelphia Electric Company proposed changes to the Technical Specifications appended to Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Units Nos. 2 and 3. Among the licensee's requests were proposed changes that would:

1. Delete the requirements for an Annual Operating Report in order to be consistent with recent Commission guidance,

2. Revise the plant organization and the membership of the Plant Operations Review Committee (PORC) to reflect the addition of a Plant Engineer - Operations to the facility staff; and

3. Revise other sections of the Technical Specifications where the changes are either editorial or pro forma in nature.

Background

After two years of experience with the reporting requirements for nuclear power reactors, we reviewed the scope of information licensees are required to submit in the Licensee Event Report (LER), Annual Operating Report, Monthly Operating Report and the Startup Report. Based on our review of LER's we developed a modified format for the LER to make this document more useful for evaluation purposes. By letters sent in July and August 1977, we informed licensees of the new LER format and requested that they use it.

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

Evaluation

1. Reporting Requirements:

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

2. Facility Staffing and PORC Membership:

   The licensee proposed to revise the plant organization for conduct of plant operations and the membership of the PORC by adding an Operations Engineer, licensed as a Senior Reactor Operator. This position has the responsibility for planning and coordinating operations in accordance with the Technical Specifications. We have reviewed the licensee's proposal and determined that the total expertice of the staff will not be reduced but will be maintained at an acceptable level. Further we have determined that the function and qualification of the PORC have not been altered by the proposed revision and therefore it is acceptable.

3. <u>Other Technical Specification Changes</u>:

The other changes proposed by the licensee include (a) rearranging in alphabetical order, the Definitions section: (b) revising certain surveillance instrument titles and ranges to be consistent with as-installed equipment and (c) permitting the PORC fourteen days (in lieu of seven) to review temporary changes to procedures. Additionally, we have included revisions to the bases portion of certain previously issued amendments that contained typographical errors. These were discussed with the licensee and he agrees. Since the above described changes are administrative in nature, do not involve changes to operating limits or surveillance requirements and are consistent with our guidance as issued in Standard Technical Specifications, we find them acceptable.

Environmental Consideration

We have determined that this amendment does not authorize a change in effluent types or total amount or an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

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

Dated: December 13, 1977

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

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 37 and 37 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 Nos. 2 and 3, located in Peach Bottom, York County, Pennsylvania. The amendments are effective as of the date of issuance.

These amendments will (1) delete the requirement for an Annual Operating Report in order to be consistent with Commission guidance; (2) revise the plant organization and the membership of the Plant Operation Review Committee (PORC) to reflect the addition of a Plant Engineer - Operations to the facility staff; and (3) revise other sections of the Technical Specifications where the changes are either editorial or pro forma in nature.

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

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the applications for amendments dated August 26, 1976, May 6, 1977, September 21, 1977, and October 31, 1977, (2) Amendments No. 37 and 37 to Licenses DPR-44 and DPR-56, 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 Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 13 day of December 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

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