

June 18, 1996

Mr. George A. Hunger,
Director-Licensing, MC 62A-1
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, PA 19087-0195

SUBJECT: TECHNICAL SPECIFICATIONS REGARDING 10 CFR 50, APPENDIX J, OPTION B,
CONTAINMENT LEAKAGE RATE TESTING REQUIREMENTS, PEACH BOTTOM ATOMIC
POWER STATION, UNIT NOS. 2 AND 3 (TAC NOS. M94853 AND M94854)

Dear Mr. Hunger:

The Commission has issued the enclosed Amendments Nos. 214 and 219 to Facility
Operating License Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power
Station, Unit Nos. 2 and 3. These amendments consist of changes to the
Technical Specifications in response to your application dated February 15,
1996.

These amendments changes the Technical Specifications to implement 10 CFR Part
50, Appendix J, Option B, by creating Technical Specification Section 5.5.12,
"Primary Containment Leakage Rate Testing Program," which refers to Regulatory
Guide 1.163, "Performance-Based Containment Leak-Test Program."

You are requested to inform the staff when you have implemented the provisions
of these amendments.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be
included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/s/

Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

240083

Docket Nos. 50-277/278

Enclosures: 1. Amendment No. 214 to DPR-44
2. Amendment No. 219 to DPR-56
3. Safety Evaluation

cc w/encls: See next page

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* Previous concurrence

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

June 18, 1996

Mr. George A. Hunger, Jr.
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PECO Energy Company
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POWER STATION, UNIT NOS. 2 AND 3 (TAC NOS. M94853 AND M94854)

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You are requested to inform the staff when you have implemented the provisions of these amendments.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

A handwritten signature in black ink, appearing to read "J. W. Shea", is written over the typed name.

Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-277/278

Enclosures: 1. Amendment No. 214 to DPR-44
2. Amendment No. 219 to DPR-56
3. Safety Evaluation

cc w/encls: See next page

Mr. George A. Hunger, Jr.
PECO Energy Company

Peach Bottom Atomic Power Station,
Units 2 and 3

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

PECO ENERGY 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. 214
License No. DPR-44

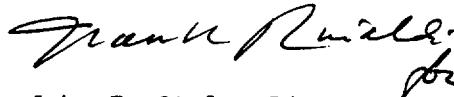
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by PECO Energy Company, et al. (the licensee) dated February 15, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - 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. 214, are hereby incorporated in the license. PECO Energy Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and is to be implemented by June 28, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment: Changes to the Technical
Specifications

Date of Issuance: June 18, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 214

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 areas are indicated by marginal lines.

<u>Remove</u>	<u>Insert</u>
1.1-3	1.1-3
3.6-2	3.6-2
3.6-6	3.6-6
3.6-16	3.6-16
5.0-17	5.0-17
5.0-18	5.0-18
5.0-19	5.0-19
5.0-20	5.0-20
5.0-21	5.0-21
5.0-22	5.0-22
5.0-23	5.0-23
5.0-24	5.0-24
5.0-25	5.0-25
-	5.0-26
B 3.0-12	B 3.0-12
B 3.6-1	B 3.6-1
B 3.6-2	B 3.6-2
B 3.6-3	B 3.6-3
B 3.6-4	B 3.6-4
B 3.6-5	B 3.6-5
B 3.6-11	B 3.6-11
B 3.6-12	B 3.6-12
B 3.6-13	B 3.6-13
B 3.6-29	B 3.6-29
B 3.6-30	B 3.6-30

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE;

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

LINEAR HEAT GENERATION
RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.1.1 Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>
<p>SR 3.6.1.1.2 Verify drywell to suppression chamber bypass leakage is equivalent to a hole ≤ 1.0 inches in diameter.</p>	<p>24 months</p> <p><u>AND</u></p> <p>-----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass -----</p> <p>12 months</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u>	12 hours
	D.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. <p>-----</p> <p>Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.1.3.14 Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15 Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16 Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	48 months

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or
 2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or
 3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.
- c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_d , is 49.1 psig.

The maximum allowable primary containment leakage rate, L_d , at P_d , shall be 0.5% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

- a. Primary Containment leakage rate acceptance criterion is $\leq 1.0 L_d$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_d$ for the Type B and Type C tests and $\leq 0.75 L_d$ for Type A tests;

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

b. Air lock testing acceptance criteria are:

- 1) Overall air lock leakage rate is ≤ 9000 scc/min when tested at $\geq P_1$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving an annual deep dose equivalent > 100 mrem and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescence dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report shall be submitted by March 31 of each year.

5.6.2 Annual Radiological Environmental Operating Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 31 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring activities for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
 1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;
 2. The Minimum Critical Power Ratio for Specifications 3.2.2 and 3.3.2.1;
 3. The Linear Heat Generation Rate for Specification 3.2.3; and
 4. The Control Rod Block Instrumentation for Specification 3.3.2.1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
 1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version as specified in the COLR);
 2. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 1, February, 1993;
 3. PECO-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code";
 4. PECO-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECo)";
 5. PECO-FMS-0003-A, "Steady-State Fuel Performance Methods Report";
 6. PECO-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis";

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. PECO-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"; and
 8. PECO-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations."
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Areas

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

- 5.7.1 High Radiation Areas with Dose Rates not Exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation):
- a. Each accessible entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.
 - b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
 - c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.
 - d. Each individual or group entering such an area shall possess:
 1. A radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"), or
 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint, or
 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

(continued)

5.7 High Radiation Areas

5.7.1 High Radiation Areas with Dose Rates not Exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation): (continued)

4. A direct-reading dosimeter and,
 - (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area, or
 - (b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area.
- e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

- a. Each accessible entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:
 1. All such door and gate keys shall be maintained under the administrative control of radiation protection personnel.
 2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)

5.7 High Radiation Areas

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual (whether alone or in a group) entering such an area shall possess:
 - 1. An alarming dosimeter with an appropriate alarm setpoint, or
 - 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or
 - 3. A direct-reading dosimeter and,
 - (a) Be under the surveillance, as specified in the RWP or equivalent, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area, or

(continued)

5.7 High Radiation Areas

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

- (b) Be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.
 - 4. A radiation monitoring and indicating device in those cases where the options of Specifications 5.7.2.d.2 and 5.7.2.d.3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.
 - e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.
 - f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.
-

BASES

SR 3.0.2
(continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SR include a Note in the Frequency stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Primary Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment

BASES

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Primary System following a Design Basis Accident (DBA) and to confine the postulated release of radioactive material. The primary containment consists of a steel lined, reinforced concrete vessel (for shielding purposes), which surrounds the Reactor Primary System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment.

The isolation devices for the penetrations in the primary containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

- a. All penetrations required to be closed during accident conditions are either:
 1. capable of being closed by an OPERABLE automatic Containment Isolation System, or
 2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";
- b. The primary containment air lock is OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Lock"; and
- c. All equipment hatches are closed.

This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of Reference 1. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J, Option B (Ref. 3), as modified by approved exemptions.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in Reference 1. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_d) is 0.5% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_d) of 49.1 psig. The value of P_d (49.1 psig) is conservative with respect to the current calculated peak drywell pressure of 47.2 psig (Ref. 2).

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0 L_d$, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the pressure suppression function is accomplished and the suppression chamber pressure does not exceed design limits. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

(continued)

BASES

LCO (continued)	Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.
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APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.
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ACTIONS

A.1

In the event primary containment is inoperable, primary containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimal.

B.1 and B.2

If primary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing (SR 3.6.1.2.1), or main steam isolation

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1.1 (continued)

valve leakage (SR 3.6.1.3.14), does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be $< 0.6 L_s$ for combined Type B and C leakage, and $< 0.75 L_s$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0 L_s$. At $\leq 1.0 L_s$, the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR is a leak test that confirms that the bypass area between the drywell and the suppression chamber is less than or equivalent to a one-inch diameter hole (Ref. 4). This ensures that the leakage paths that would bypass the suppression pool are within allowable limits.

The leakage test is performed every 24 months. The 24 month Frequency was developed considering that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, a test shall be performed at a Frequency of once every 12 months until two consecutive tests pass, at which time the 24 month test Frequency may be resumed.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 14.6.3.
 2. Letter G94-PEPR-183, Peach Bottom Improved Technical Specification Project Increased Drywell and Suppression Chamber Pressure Analytical Limits, from G.V. Kumar (GE) to A.A. Winter (PECO), August 23, 1994.
 3. 10 CFR 50, Appendix J, Option B.
 4. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 127 and 130 to Facility Operating License Nos. DPR-44 and DPR-56, dated February 18, 1988.
 5. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."
 6. ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements."
-

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

limits. In many instances (e.g., only one seal per door has failed), primary containment remains OPERABLE, yet only 1 hour (according to LCO 3.6.1.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with the overall air lock leakage not within limits, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the primary containment air lock must be verified closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1.1, which require that primary containment be restored to OPERABLE status within 1 hour.

Additionally, the air lock must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is reasonable for restoring an inoperable air lock to OPERABLE status considering that at least one door is maintained closed in the air lock.

D.1 and D.2

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.2.1

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with respect to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.2.1 (continued)

testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 requires the results of air lock leakage tests to be evaluated against the acceptance criteria of the Primary Containment Leakage Rate Testing Program, 5.5.12. This ensures that the air lock leakage is properly accounted for in determining the combined Type B and C primary containment leakage.

SR 3.6.1.2.2

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure, closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when primary containment is entered, this test is only required to be performed upon entering primary containment, but is not required more frequently than 184 days when primary containment is de-inerted. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other administrative controls available to operations personnel.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 5.2.3.4.5.
 2. 10 CFR 50, Appendix J, Option B.
 3. Letter G94-PEPR-183, Peach Bottom Improved Technical Specification Project Increased Drywell and Suppression Chamber Pressure Analytical Limits, from G.V. Kumar (GE) to A.A. Winter (PECo), August 23, 1994.
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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

The analyses in Reference 1 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 11.5 scfh when tested at $\geq P_c$ (25 psig). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.1.3.16

The inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve must be replaced every 48 months. This will allow the opportunity for replacement before gross leakage failure occurs. The 48 month Frequency is based on vendor recommendation and operational experience which shows that gross leakage normally does not occur when the inflatable seals are replaced on a 48 month Frequency.

REFERENCES

1. UFSAR, Chapter 14.
 2. UFSAR, Table 7.3.1.
 3. 10 CFR 50, Appendix J, Option B.
 4. UFSAR, Table 7.3.1, Note 17.
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PECO ENERGY 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. 219
License No. DPR-56

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by PECO Energy Company, et al. (the licensee) dated February 15, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health or 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. 219, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and is to be implemented by June 28, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in dark ink, appearing to read "John F. Stolz", with a stylized flourish at the end.

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

Attachment: Changes to the Technical
Specifications

Date of Issuance: June 18, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 219

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 areas are indicated by marginal lines.

<u>Remove</u>	<u>Insert</u>
1.1-3	1.1-3
3.6-2	3.6-2
3.6-6	3.6-6
3.6-16	3.6-16
5.0-17	5.0-17
5.0-18	5.0-18
5.0-19	5.0-19
5.0-20	5.0-20
5.0-21	5.0-21
5.0-22	5.0-22
5.0-23	5.0-23
5.0-24	5.0-24
5.0-25	5.0-25
-	5.0-26
B 3.0-12	B 3.0-12
B 3.6-1	B 3.6-1
B 3.6-2	B 3.6-2
B 3.6-3	B 3.6-3
B 3.6-4	B 3.6-4
B 3.6-5	B 3.6-5
B 3.6-11	B 3.6-11
B 3.6-12	B 3.6-12
B 3.6-13	B 3.6-13
B 3.6-29	B 3.6-29
B 3.6-30	B 3.6-30

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE;

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

LINEAR HEAT GENERATION
RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.1.2	Verify drywell to suppression chamber bypass leakage is equivalent to a hole ≤ 1.0 inches in diameter.	24 months <u>AND</u> -----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass ----- 12 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1.1. <p>-----</p> <p>Perform required primary containment air lock leakage rate testing in accordance with the Primary Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Primary Containment Leakage Rate Testing Program</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.1.3.14 Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15 Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16 Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	48 months

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or
 2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or
 3. A required system redundant to support system(s) for the supported systems (b.1) and (b.2) above is also inoperable.
- c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_s , is 49.1 psig.

The maximum allowable primary containment leakage rate, L_s , at P_s , shall be 0.5% of primary containment air weight per day.

Leakage Rate acceptance criteria are:

- a. Primary Containment leakage rate acceptance criterion is $\leq 1.0 L_s$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_s$ for the Type B and Type C tests and $\leq 0.75 L_s$ for Type A tests;

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

b. Air lock testing acceptance criteria are:

- 1) Overall air lock leakage rate is ≤ 9000 scc/min when tested at $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving an annual deep dose equivalent > 100 mrem and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescence dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report shall be submitted by March 31 of each year.

5.6.2 Annual Radiological Environmental Operating Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 31 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring activities for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
 1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;
 2. The Minimum Critical Power Ratio for Specifications 3.2.2 and 3.3.2.1;
 3. The Linear Heat Generation Rate for Specification 3.2.3; and
 4. The Control Rod Block Instrumentation for Specification 3.3.2.1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
 1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version as specified in the COLR);
 2. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 1, February, 1993;
 3. PECO-FMS-0001-A, "Steady-State Thermal Hydraulic Analysis of Peach Bottom Units 2 and 3 using the FIBWR Computer Code";
 4. PECO-FMS-0002-A, "Method for Calculating Transient Critical Power Ratios for Boiling Water Reactors (RETRAN-TCPPECO)";
 5. PECO-FMS-0003-A, "Steady-State Fuel Performance Methods Report";
 6. PECO-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis";

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. PECO-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"; and
 8. PECO-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations."
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Areas

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

- 5.7.1 High Radiation Areas with Dose Rates not Exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation):
- a. Each accessible entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.
 - b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
 - c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.
 - d. Each individual or group entering such an area shall possess:
 1. A radiation monitoring device that continuously displays radiation dose rates in the area ("radiation monitoring and indicating device"), or
 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached ("alarming dosimeter"), with an appropriate alarm setpoint, or
 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

(continued)

5.7 High Radiation Areas

5.7.1 High Radiation Areas with Dose Rates not Exceeding 1.0 rem/hour (at 30 centimeters from the radiation sources or from any surface penetrated by the radiation): (continued)

4. A direct-reading dosimeter and,
 - (a) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual at the work site, qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel radiation exposure within the area, or
 - (b) Be under the surveillance, as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area.
- e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation)

- a. Each accessible entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked door, gate, or guard that prevents unauthorized entry, and in addition:
 1. All such door and gate keys shall be maintained under the administrative control of radiation protection personnel.
 2. Doors and gates shall remain locked or guarded except during periods of personnel or equipment entry or exit.

(continued)

5.7 High Radiation Areas

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual (whether alone or in a group) entering such an area shall possess:
 - 1. An alarming dosimeter with an appropriate alarm setpoint, or
 - 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or
 - 3. A direct-reading dosimeter and,
 - (a) Be under the surveillance, as specified in the RWP or equivalent, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring and indicating device who is responsible for controlling personnel exposure within the area, or

(continued)

5.7 High Radiation Areas

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour (at 30 centimeters from the radiation source or from any surface penetrated by the radiation), but less than 500 rads/hour (at 1 meter from the radiation source or from any surface penetrated by the radiation) (continued)

- (b) Be under the surveillance, as specified in the RWP or equivalent, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.
 - 4. A radiation monitoring and indicating device in those cases where the options of Specifications 5.7.2.d.2 and 5.7.2.d.3, above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle.
 - e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been established and entry personnel are knowledgeable of them.
 - f. Such individual areas that are within a larger area that is controlled as a high radiation area, where no enclosure exists for purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, but shall be barricaded and conspicuously posted as a high radiation area, and a conspicuous, clearly visible flashing light shall be activated at the area as a warning device.
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BASES

SR 3.0.2
(continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the SR include a Note in the Frequency stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Primary Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment

BASES

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Primary System following a Design Basis Accident (DBA) and to confine the postulated release of radioactive material. The primary containment consists of a steel lined, reinforced concrete vessel (for shielding purposes), which surrounds the Reactor Primary System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment.

The isolation devices for the penetrations in the primary containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

- a. All penetrations required to be closed during accident conditions are either:
 1. capable of being closed by an OPERABLE automatic Containment Isolation System, or
 2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";
- b. The primary containment air lock is OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Lock"; and
- c. All equipment hatches are closed.

This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of Reference 1. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J, Option B (Ref. 3), as modified by approved exemptions.

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in Reference 1. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_p) is 0.5% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_p) of 49.1 psig. The value of P_p (49.1 psig) is conservative with respect to the current calculated peak drywell pressure of 47.2 psig (Ref. 2).

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0 L_p$, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the pressure suppression function is accomplished and the suppression chamber pressure does not exceed design limits. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

(continued)

BASES

LCO (continued)	Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.
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APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.
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ACTIONS

A.1

In the event primary containment is inoperable, primary containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining primary containment OPERABILITY during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimal.

B.1 and B.2

If primary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing (SR 3.6.1.2.1), or main steam isolation

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1.1 (continued)

valve leakage (SR 3.6.1.3.14), does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. As left leakage prior to the first startup after performing a required leakage test is required to be $< 0.6 L_s$ for combined Type B and C leakage, and $< 0.75 L_s$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0 L_s$. At $\leq 1.0 L_s$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.1.2

Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR is a leak test that confirms that the bypass area between the drywell and the suppression chamber is less than or equivalent to a one-inch diameter hole (Ref. 4). This ensures that the leakage paths that would bypass the suppression pool are within allowable limits.

The leakage test is performed every 24 months. The 24 month Frequency was developed considering that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, a test shall be performed at a Frequency of once every 12 months until two consecutive tests pass, at which time the 24 month test Frequency may be resumed.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 14.6.3.
 2. Letter G94-PEPR-183, Peach Bottom Improved Technical Specification Project Increased Drywell and Suppression Chamber Pressure Analytical Limits, from G.V. Kumar (GE) to A.A. Winter (PECO), August 23, 1994.
 3. 10 CFR 50, Appendix J, Option B.
 4. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 127 and 130 to Facility Operating License Nos. DPR-44 and DPR-56, dated February 18, 1988.
 5. NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."
 6. ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements."
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BASES

ACTIONS

C.1, C.2, and C.3 (continued)

limits. In many instances (e.g., only one seal per door has failed), primary containment remains OPERABLE, yet only 1 hour (according to LCO 3.6.1.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with the overall air lock leakage not within limits, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the primary containment air lock must be verified closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1.1, which require that primary containment be restored to OPERABLE status within 1 hour.

Additionally, the air lock must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is reasonable for restoring an inoperable air lock to OPERABLE status considering that at least one door is maintained closed in the air lock.

D.1 and D.2

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.2.1

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with respect to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.2.1 (continued)

testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 requires the results of air lock leakage tests to be evaluated against the acceptance criteria of the Primary Containment Leakage Rate Testing Program, 5.5.12. This ensures that the air lock leakage is properly accounted for in determining the combined Type B and C primary containment leakage.

SR 3.6.1.2.2

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure, closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when primary containment is entered, this test is only required to be performed upon entering primary containment, but is not required more frequently than 184 days when primary containment is de-inerted. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other administrative controls available to operations personnel.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 5.2.3.4.5.
 2. 10 CFR 50, Appendix J, Option B.
 3. Letter G94-PEPR-183, Peach Bottom Improved Technical Specification Project Increased Drywell and Suppression Chamber Pressure Analytical Limits, from G.V. Kumar (GE) to A.A. Winter (PECo), August 23, 1994.
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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

The analyses in Reference 1 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 11.5 scfh when tested at $\geq P_t$ (25 psig). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.1.3.16

The inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve must be replaced every 48 months. This will allow the opportunity for replacement before gross leakage failure occurs. The 48 month Frequency is based on vendor recommendation and operational experience which shows that gross leakage normally does not occur when the inflatable seals are replaced on a 48 month Frequency.

REFERENCES

1. UFSAR, Chapter 14.
 2. UFSAR, Table 7.3.1.
 3. 10 CFR 50, Appendix J, Option B.
 4. UFSAR, Table 7.3.1, Note 17.
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NOS. 214 AND 219 TO FACILITY OPERATING

LICENSE NOS. DPR-44 and DPR-56

PECO ENERGY COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B "Performance-Based Requirements" to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By application dated February 15, 1996, PECO Energy Company, (PECO, the licensee) requested changes to the Technical Specification (TS) for Peach Bottom Atomic Power Station, Units 2 and 3, (PBAPS). The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B. The licensee has established a "Containment Leakage Rate Testing Program" and proposed adding this program to the TS. The program references Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak-Test Program" which specifies a method acceptable to the NRC for complying with Option B.

Proposed TS page 5.0-17a of the February 15, 1996 application was renumbered as page 5.0-18 before issuing this amendment and existing TS pages 5.0-18 through 5.0-25 are being reissued as pages 5.0-19 through 5.0-26.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50, "Primary Containment Leakage Testing for Water-Cooled Power Reactors" was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, September 1995, "Performance-Based Containment Leak-Test Program", was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced Regulatory Guide 1.163 in the PBAPS TS.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were transmitted to NEI in a letter from C. Grimes, NRC, to D. Modeen, NEI, dated November 2, 1995. These TS are to serve as a model for licensees to develop plant specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these

limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's February 15, 1996, letter to the NRC proposes to establish a "Containment Leakage Rate Testing Program" and proposes to add this program to the TS. The program references Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program" which specifies a method acceptable to the NRC for complying with Option B. This requires a change to:

(1) Existing TS 1.1 "Definitions" - deletion of the definition of L_a

(2) Existing TS 3.6.1.1, "Primary Containment,"
TS 3.6.1.2, "Primary Containment Air Lock,"
TS 3.6.1.3, "PCIVs,"

- deletion of specific acceptance criteria and surveillance frequency and insertion of reference to "Primary Containment Leakage Rate Testing Program"

(3) Existing TS 5.5, "Programs and Manuals" - addition of new TS 5.5.12, defining the "Primary Containment Leakage Rate Testing Program"

Corresponding bases were also modified.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis.

The TS changes proposed by the licensee are in compliance with the requirements of 10 CFR Part 50, Appendix J, Option B and consistent with the guidance of Regulatory Guide 1.163, September 1995, and the generic TS of the November 2, 1995, letter and are, therefore, acceptable to the staff.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (61 FR 13531). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributor: J. Shea

Date: June 18, 1996