November 19, 1985

Dockets Nos. 50-277
and 50-278

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

Dear Mr. Bauer:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENTS PERTAINING TO REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION

Re: Peach Bottom Atomic Power Station, Units Nos. 2 and 3.

The Commission has issued the enclosed Amendments Nos. 112 and 116, 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 (TSs) in response to your application dated May 4, 1983, as amended by letters dated November 10, 1983, and November 29, 1984.

The changes involve revisions in the Reactor Coolant Pressure Boundary leak detection TSs pertaining to the measuring of airborne radioactivity in the primary containment. The Commission gave its previous approval to your other amendment requests contained in the May 4, 1983, and November 10, 1983, submittals by letter dated February 27, 1985. This completes our response to the above cited submittals.

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

Sincerely,

Original signed by

Gerald Gears, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 112 to DPR-44
2. Amendment No. 116 to DPR-56
3. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:ORB#4:DL
RIngram	GGears;cr	JStolz	LFinkelstein	GLainas
11/5/85	11/6/85	11/8/85	11/13/85	11/19/85

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PDR ADOCK 05000277
P PDR

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

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

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated May 4, 1983, as amended by letters dated November 10, 1983, and November 29, 1984, 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 112, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

A E Edison for

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 19, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 112

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 a vertical line indicating the area of change.

Remove	Insert
v	v
76	76
85	85
93	93
146	146
146a	146a
156	156
156a	156a

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3.2.B	Instrumentation That Initiates or Controls the Core and Containment Cooling Systems	64
3.2.C	Instrumentation That Initiates Control Rod Blocks	73
3.2.D	Radiation Monitoring Systems That Initiate and/or Isolates Systems	75
3.2.F	Surveillance Instrumentation	77
3.2.G	Instrumentation That Initiates Recirculation Pump Trip	79
4.2.A	Minimum Test and Calibration Frequency for PCIS	80

Table 3.2.E Deleted

Amendment No. 112

TABLE 4.2.E

MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Equipment Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
2) Floor Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
3) Drywell Atmosphere Radioactivity Monitor	(1)	Once/3 months	Once/day

3.2 BASES (Cont'd)

Four sets of two radiation monitors are provided which initiate the Reactor Building Isolation function and operation of the standby gas treatment system. Four instrument channels monitor the radiation from the refueling area ventilation exhaust ducts and four instrument channels monitor the building ventilation below the refueling floor. Each set of the instrument channels is arranged in a 1 out of 2 twice trip logic.

Trip settings of <16 mr/hr for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

Flow integrators are used to record the integrated flow of liquid from the drywell sumps. The integrated flow is indicative of reactor coolant leakage. A Drywell Atmosphere Radioactivity Monitor is provided to give supporting information to that supplied by the reactor coolant leakage monitoring system. (See Bases for 3.6.C and 4.6.C)

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event fall within the envelope of study events given in General Electric Company Topical Report, NEDO-10439, dated March, 1971.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.B Coolant Chemistry (Cont'd)	4.6.B Coolant Chemistry (Cont'd)
b) Chloride Concentration Time above 2 weeks/year 0.2 ppm Maximum limit 1.0 ppm	
c) pH During operations, if the conductivity exceeds 1.0 umho/cm, pH shall be measured and brought within the 5.6 to 8.6 range within 24 hours. If the pH cannot be corrected, or if the pH is outside a range of 4 to 10, the unit shall be placed in Hot Shutdown within 12 hours and in Cold Shutdown with 36 hours.	
C. Coolant Leakage	C. Coolant Leakage
1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212 degrees F, the rate of reactor coolant leakage to the primary containment from unidentified sources shall not exceed 5 gallons per minute. The rate of change of unidentified leakage shall not exceed 2 gallons per minute per 24 hour surveillance period when the reactor is operated in the "Run" mode. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm averaged over any 24 hour surveillance period.	1. Reactor coolant system leakage shall be determined by the primary containment (Drywell) sump collection and flow monitoring system and recorded every 4 hours or less. 2. Drywell atmosphere radioactivity levels shall be monitored and recorded at least once per day.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.6.C. Coolant Leakage

2. The primary containment (Drywell) sump collection and flow monitoring system shall be operable during reactor power operation. From and after the time that this system is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 24 hours unless the system is made operable sooner. For purposes of this paragraph, the primary containment (Drywell) sump collection and flow monitoring system operability is defined as the ability to measure reactor coolant leakage.

3. The Drywell Atmosphere Radioactivity Monitor shall be operable during reactor power operation as a supplement to the reactor coolant leakage monitoring system. From and after the time that this system is made or found to be inoperable for any reason, reactor power operation is permissible for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours.

4. If the conditions in 1, 2, or 3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown Condition within the following 24 hours.

3.6.C & 4.6.C BASES

Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite ac power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the plant should be shutdown to allow further investigation and corrective action.

A rate of change limit of 2 gpm per 24 hour surveillance period is specified to provide additional conservatism. This limit is applicable to reactor operations in the "Run" mode, during which time there is little variation in primary coolant system pressure. The limit does not apply to the "Startup" mode since this period is characterized by large variations in system pressure and consequently, changes in measured leakage would not be indicative of system degradation. During the limited duration of the startup phase, the 5 gpm limit will ensure the integrity of the primary coolant system.

The total leakage rate consists of all leakage, unidentified and identified, which flows to the drywell floor drain and equipment drain sumps, respectively. Both the Drywell floor drain and the equipment drain sumps have pump-out capacitites of 50 gpm per pump. Any one pump can therefore handle in excess of the maximum allowable total leakage of 25 gpm. If the ability to measure pump-out flow from either of these sumps is lost, the inoperable

sump will overflow into the remaining operable sump. The remaining operable sump pump-out flow will then represent the total leakage rate. During the time when one sump is overflowing, any increase in total flow will be assumed to be from an unidentified source. This primary containment (Drywell) sump collection and flow monitoring system can provide viable measurement of reactor coolant system leakage so long as one pump and its associated flow meter are operable.

The Drywell Atmosphere Radioactivity Monitor provides supporting information to that provided by the reactor coolant leakage monitoring system. There is no direct correlation between the radioactivity monitor indication and the leakage rate because of the uncertainties regarding coolant activity levels, source of leakage, and background radiation levels. While the radioactivity monitors will not quantify primary coolant leakage, they would provide an early warning of a major leak especially if there is a significant difference in the radioactivity level between the leakage source and drywell background.

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. 116
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated May 4, 1983, as amended by letters dated November 10, 1983, and November 29, 1984, 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-56 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.116 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

A E Edison for

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 19, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 116

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 a vertical line indicating the area of change.

Remove	Insert
v	v
76	76
85	85
93	93
146	146
146a	146a
156	156
156a	156a

LIST OF TABLES

Table	Title	Page
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3.2.C	Instrumentation That Initiates Control Rod Blocks	73
3.2.D	Padiation Monitoring Systems That Initiate and/or Isolates Systems	75
3.2.F	Surveillance Instrumentation	77
3.2.G	Instrumentation That Initiates Recirculation Pump Trip	79
4.2.A	Minimum Test and Calibration Frequency for PCIS	80

Table 3.2.E Deleted

Amendment No. 116

TABLE 4.2.E

MINIMUM TEST AND CALIBRATION FREQUENCY FOR DRYWELL LEAK DETECTION

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Equipment Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
2) Floor Drain Sump Flow Integrator	(1)	Once/3 months	Once/day
3) Drywell Atmosphere Radioactivity Monitor	(1)	Once/3 months	Once/day

3.2 BASES (Cont'd)

Four sets of two radiation monitors are provided which initiate the Reactor Building Isolation function and operation of the standby gas treatment system. Four instrument channels monitor the radiation from the refueling area ventilation exhaust ducts and four instrument channels monitor the building ventilation below the refueling floor. Each set of the instrument channels is arranged in a 1 out of 2 twice trip logic.

Trip settings of <16 mr/hr for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

Flow integrators are used to record the integrated flow of liquid from the drywell sumps. The integrated flow is indicative of reactor coolant leakage. A Drywell Atmosphere Radioactivity Monitor is provided to give supporting information to that supplied by the reactor coolant leakage monitoring system. (See Bases for 3.6.C and 4.6.C)

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event fall within the envelope of study events given in General Electric Company Topical Report, NEDO-10439, dated March, 1971.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.B Coolant Chemistry (Cont'd)	4.6.B Coolant Chemistry (Cont'd)
b) Chloride Concentration Time above 2 weeks/year 0.2 ppm Maximum limit 1.0 ppm	
c) pH During operations, if the conductivity exceeds 1.0 umho/cm, pH shall be measured and brought within the 5.6 to 8.6 range within 24 hours. If the pH cannot be corrected, or if the pH is outside a range of 4 to 10, the unit shall be placed in Hot Shutdown within 12 hours and in Cold Shutdown with 36 hours.	
C. Coolant Leakage	C. Coolant Leakage
1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212 degrees F, the rate of reactor coolant leakage to the primary containment from unidentified sources shall not exceed 5 gallons per minute. The rate of change of unidentified leakage shall not exceed 2 gallons per minute per 24 hour surveillance period when the reactor is operated in the "Run" mode. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm averaged over any 24 hour surveillance period.	1. Reactor coolant system leakage shall be determined by the primary containment (Drywell) sump collection and flow monitoring system and recorded every 4 hours or less. 2. Drywell atmosphere radioactivity levels shall be monitored and recorded at least once per day.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

3.6.C. Coolant Leakage

2. The primary containment (Drywell) sump collection and flow monitoring system shall be operable during reactor power operation. From and after the time that this system is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 24 hours unless the system is made operable sooner. For purposes of this paragraph, the primary containment (Drywell) sump collection and flow monitoring system operability is defined as the ability to measure reactor coolant leakage.

3. The Drywell Atmosphere Radioactivity Monitor shall be operable during reactor power operation as a supplement to the reactor coolant leakage monitoring system. From and after the time that this system is made or found to be inoperable for any reason, reactor power operation is permissible for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours.

4. If the conditions in 1, 2, or 3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown Condition within the following 24 hours.

3.6.C & 4.6.C BASES

Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite ac power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the plant should be shutdown to allow further investigation and corrective action.

A rate of change limit of 2 gpm per 24 hour surveillance period is specified to provide additional conservatism. This limit is applicable to reactor operations in the "Run" mode, during which time there is little variation in primary coolant system pressure. The limit does not apply to the "Startup" mode since this period is characterized by large variations in system pressure and consequently, changes in measured leakage would not be indicative of system degradation. During the limited duration of the startup phase, the 5 gpm limit will ensure the integrity of the primary coolant system.

The total leakage rate consists of all leakage, unidentified and identified, which flows to the drywell floor drain and equipment drain sumps, respectively. Both the Drywell floor drain and the equipment drain sumps have pump-out capacitites of 50 gpm per pump. Any one pump can therefore handle in excess of the maximum allowable total leakage of 25 gpm. If the ability to measure pump-out flow from either of these sumps is lost, the inoperable

sump will overflow into the remaining operable sump. The remaining operable sump pump-out flow will then represent the total leakage rate. During the time when one sump is overflowing, any increase in total flow will be assumed to be from an unidentified source. This primary containment (Drywell) sump collection and flow monitoring system can provide viable measurement of reactor coolant system leakage so long as one pump and its associated flow meter are operable.

The Drywell Atmosphere Radioactivity Monitor provides supporting information to that provided by the reactor coolant leakage monitoring system. There is no direct correlation between the radioactivity monitor indication and the leakage rate because of the uncertainties regarding coolant activity levels, source of leakage, and background radiation levels. While the radioactivity monitors will not quantify primary coolant leakage, they would provide an early warning of a major leak especially if there is a significant difference in the radioactivity level between the leakage source and drywell background.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENTS NOS. 112 AND 116 TO FACILITY OPERATING LICENSES NOS. DPR-44 AND DPR-56

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

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

DOCKETS NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By letter dated May 4, 1983, and amended by letters dated November 10, 1983, and November 29, 1984, Philadelphia Electric Company (the licensee) made application to amend the Technical Specifications (TSs) for the Peach Bottom Atomic Power Station, Units 2 and 3, to permit revisions in the Reactor Coolant Pressure Boundary (RCPB) leak detection TSs (Sections 3.6.C and 4.6.C and Tables 3.2.E and 4.2.E). By letter dated February 27, 1985, the Commission approved certain other requested changes proposed in the licensee's submittals dated May 4, 1983, and November 10, 1983. This Safety Evaluation addresses the remaining open items identified in our February 27, 1985, letter to the licensee and specifically addresses the licensee's November 29, 1984, submittal.

2.0 EVALUATION

Specifically, the licensee's applications requested the following: (1) a change in nomenclature from "Air Sampling System" to "Drywell Atmosphere Radioactivity Monitor" to clarify the parameter being monitored; (2) elimination of Table 3.2.E (Instrumentation That Monitors Drywell Leak Detection) to remove redundancy in the TSs; (3) a change in limiting condition for operation (LCO) requirements from a 7-day to a 30-day LCO together with the addition of grab sample surveillance requirements in the revised Section 3.6.C.3; and (4) revisions to TS Bases supporting these changes. The November 29, 1984 letter also withdraws certain requests involving the proposed deletion of specific testing and surveillance requirements concerning the Drywell Atmosphere Radioactivity Monitor Systems.

2.1.1 Nomenclature Change

The licensee has changed the name of the system measuring airborne radioactivity in the primary containment from "Air Sampling System" to "Drywell Atmosphere Radioactivity Monitor" in order to relate to the parameter being monitored. We find this acceptable since the new name covers the same devices as the old one, the radioactive noble gas monitor, the iodine monitoring system and the particulate monitoring system.

2.1.2 Table 4.2.E, "Minimum Test and Calibration Frequency for Drywell Leak Detection"

The licensee has also changed the nomenclature in Table 4.2.E in accordance with the name change noted above in Item 2.1.1. We find this acceptable.

2.2 Elimination of Table 3.2.E (Instrumentation That Monitors Drywell Leak Detection)

The licensee has proposed certain revisions to Section 3.6.C (refer to Items 2.3.1 and 2.3.2 below) pertaining to the operability requirements of the Drywell Leak Detection Systems (i.e., the Drywell Sump Collection and Flow Monitoring System and the Drywell Atmosphere Radioactivity Monitor). Table 3.2.E (Instrumentation That Monitors Drywell Leak Detection) also specifies operability requirements for the same systems. The licensee proposes that Table 3.2.E be deleted because the proposed changes to Section 3.6.C will now incorporate all NRC staff approved operability requirements. We concur with the licensee's proposed change and conclude that this deletion simply eliminates a redundant section from the Peach Bottom TSs (Table 3.2.E) and, therefore, would not change the operability requirements for the Drywell Leak Detection Systems. We, therefore, find this change acceptable.

2.3.1 Limiting Conditions for Operation (LCO) - Technical Specification 3.6.C.3

The present second paragraph of LCO 3.6.C.2 concerning the air sampling system has been modified and changed to LCO 3.6.C.3. Another modification consists of a change in name to Drywell Atmosphere Radioactivity Monitor (DARM) and to indicate that reactor power operation is permitted for up to 30 days in the event the DARM is inoperable, in lieu of seven days previously allowed for inoperability of the air sampling system. However, the new specification has added grab sampling of the containment atmosphere and analysis thereof at least once every 24 hours while the DARM is inoperable. We find this acceptable because it conforms with the requirements of the LCO (3.4.3.1) for Reactor Coolant System Leakage in the Standard Technical Specifications for Boiling Water Reactors (NUREG-0123, Rev. 3).

2.3.2 Limiting Conditions for Operation (LCO) - Technical Specifications 3.6.C.4

This proposed revision would simply change the old LCO 3.6.C.3 to LCO 3.6.C.4, retaining it in its entirety, and adding a reference to LCO 3.6.C.3. This new LCO states that if excessive leakage is found beyond allowable limits shown in LCO 3.6.C.1 or that in the loss of Sump Collection and Flow Monitoring System inoperability in excess of 24 hours (LCO 3.6.C.2) or in the event of DARM inoperability in excess of 30 days, that the plant be in hot shutdown within 12 hours and in cold shutdown within 24 hours thereafter. We find this acceptable.

2.4.1 Bases, Page 93, Paragraph 3

Paragraph 3 has been modified to change the name of the air sampling system to Drywell Atmosphere Radioactivity Monitor (DARM) and to indicate that the DARM will provide information to the reactor coolant leakage detection (primary containment sump collection and flow monitoring) system. The licensee has also deleted the reference to the alarm unit in each sump flow integrator which is set to annunciate when the limiting rates of flow set by Technical Specification 3.6.C are exceeded. The Final Safety Analysis Report (FSAR), in Section 4.10.3.2.IV, notes that each sump (drywell floor drain for unidentified flow, drywell equipment drain sump for identified leakage) has a 500 gallon capacity with two pumps each capable of pumping 50 gpm. The second pump is initiated if the pump water level continues to rise above a certain point at which time an alarm is sounded. The FSAR notes, further, that a flow integrator meters the discharge rate from each sump. An operator monitors the discharge every four hours and verifies that floor drain flow has not exceeded 5 gpm, that floor drain plus equipment drain sump flow has not exceeded 25 gpm, that the increase in floor drain sump flow is less than 2 gpm, and that floor drain leakage was not doubled since the previous reading. A plant shutdown is initiated if any of these limits are exceeded. We find this acceptable.

2.4.2 Bases, Page 156a

A paragraph has been added, noting that the DARM is intended to provide information in support of that provided by the reactor coolant leakage monitoring system (sump collection and flow monitoring system), stating "There is no direct correlation between the radioactivity monitor indication and leakage rate .... While the radioactivity monitors will not quantify primary coolant leakage, they would provide an early warning of a major leak especially if there is a significant difference in the radioactivity level between the leakage source and drywell background."

We find the licensee's proposed changes to the Bases section acceptable.

2.5 Redistribution of Material (Section 4.6.C)

The second paragraph of present surveillance requirement 4.6.C.1 pertains to checking the air sampling system and recording reactor coolant system leakage as measured by the air sampling system at least once per day. This paragraph has been deleted and these requirements have been redistributed into a new section (4.6.C.2). We find this change acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these 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 these amendments.

4.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 19, 1985

The following NRC personnel have contributed to this Safety Evaluation: N. Wagner and G. Gears