November 14, 1984

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Docket No. 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: PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3, TECHNICAL SPECIFICATION AMENDMENT PERTAINING TO A LICENSE AMENDMENT APPLICATION DATED SEPTEMBER 28, 1984

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. DPR-56 for the Peach Bottom Atomic Power Station, Unit No. 3. This amendment consists of changes to the Technical Specifications (TSs) in partial response to your application dated September 28, 1984. Your request for a similar TS change for Unit 2 awaits notification by your staff of a need for a short-term hydrogen injection test at Unit 2 upon startup of that Unit in 1985. Therefore, the Unit 2 request will be the subject of a future NRC action.

The changes to the TSs permit a temporary increase in the Main Steam Line High Radiation scram and isolation setpoints to facilitate operation with the expected increased N-16 radiation levels due to a proposed short-term hydrogen injection test. This scheduled short-term hydrogen injection test at Unit 3 will be used to determine the feasibility of using hydrogen water chemistry as a means of reducing intergranular stress corrosion cracking (IGSCC) of stainless steel BWR piping.

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

Sincerely,

"ORIGINAL SIGNED BY"
Gerald E. Gears, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 106 to DPR-56
2. Safety Evaluation

cc w/enclosures:
See next page

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Philadelphia Electric Company

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

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

DOCKET NO. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
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 September 28, 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. 106, 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

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 14, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 106

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
38	38
40	40
61	61
63	63

Table 3.1.1 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in which Function Must be Operable			Number of Instrument Channels Provided by Desgin	Action (1)
			Refuel (7)	Startup	Run		
2	High Water Level in Scram Discharge Volume	≤50 Gallons	X(2)	X	X	4 Instrument Channels	A
2	Turbine Condenser Low Vacuum	≥23 in. Hg. Vacuum	X(3)	X(3)	X	4 Instrument Channels	A or C
2	Main Steam Line High Radiation	≤3 X Normal Full Power Background	X	X	X(14)	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤10% Valve Closure	X(3)(6)	X(3)(6)	X(6)	8 Instrument Channels	A
2	Turbine Control Valve Fast Closure	500<P<850 psig Control Oil Pressure Between Fast Closure Solenoid and Disc Dump Valve			X(4)	4 Instrument Channels	A or D
4	Turbine Stop Valve Closure	≤10% Valve Closure			X(4)	8 Instrument Channels	A or D

NOTES FOR TABLE 3.1.1 (Cont'd)

10. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not high.

11. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 14 LPRM inputs of the normal complement.

12. This equation will be used in the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), where:

    FRP = fraction of rated thermal power (3293 MWt).
    MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for all 8 x 8 fuel.

    The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

    W = Loop Recirculation flow in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

    Delta W = the difference between two loop and single loop effective recirculation drive flow rate at the same core flow. During single loop operation, the reduction in trip setting (-0.66 delta W) is accomplished by correcting the flow input of the flow biased High Flux trip setting to preserve the original (two loop) relationship between APRM High Flux setpoint and recirculation drive flow or by adjusting the APRM Flux trip setting. Delta W equals zero for two loop operation.

    Trip level setting is in percent of rated power (3293 MWt).

13. See Section 2.1.A.1.

14. Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full power background radiation level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of the test program, and within 12 hours of establishing reactor power levels below 20% rated power.

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum No. of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action (2)
2 (6)	Reactor Low Water Level	$\geq$ 0" Indicated Level (3)	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	$\leq$ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low Water Level	at or above -49" indicated level (4)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	$\leq$ 2 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	$\leq$ 3 X Normal Rated Full Power Background (8) (10)	4 Inst. Channels	B
2	Low Pressure Main Steam Line	$\geq$ 850 psig (7)	4 Inst. Channels	B
2 (5)	High Flow Main Steam Line	$\leq$ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Tunnel Exhaust Duct High Temperature	$\leq$ 200 deg. F (9)	4 Inst. Channels	B

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. If the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken:

   A. Initiate an orderly shutdown and have the reactor in Cold Shutdown Condition in 24 hours.

   B. Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate Shutdown Cooling.

3. Instrument setpoint corresponds to 177.7" above top of active fuel.

4. Instrument setpoint corresponds to 129.7" above top of active fuel.

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in Run Mode (interlocked with Mode Switch).

8. At a radiation level of 1.5 times the normal rated power background, an alarm will be tripped in the control room to alert the control room operators to an increase in the main steam line tunnel radiation level.

9. In the event of a loss of ventilation in the main steam line tunnel area, the main steam line tunnel exhaust duct high temperature setpoint may be raised up to 250 degrees F for a period not to exceed 30 minutes to permit restoration of the ventilation flow. During the 30-minute period, an operator shall observe control room indications of the duct temperature so in the event of rapid increases (indicative of a steam line break) the operator shall promptly close the main steam line isolation valves.

10. Within 24 hours prior to the planned start of the hydrogen injection test with the reactor power at greater than 20% rated power, the normal full power radiation background level and associated trip setpoints may be changed based on a calculated value of the radiation level expected during the test. The background radiation level and associated trip setpoints may be adjusted during the test program based on either calculations or measurements of actual radiation levels resulting from hydrogen injection. The background radiation level shall be determined and associated trip setpoints shall be set within 24 hours of re-establishing normal radiation levels after completion of the test program, and within 12 hours of establishing reactor power levels below 20% rated power.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. 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, UNIT NO. 3

DOCKET NO. 50-278

Introduction

By letter dated September 28, 1984, the Philadelphia Electric Company, et al. (the licensee) made application to amend the Technical Specifications (TSs) for the Peach Bottom Atomic Power Station, Unit No. 3, to permit a temporary increase in the Main Steam Line High Radiation scram and isolation setpoints to facilitate the short-term testing of hydrogen addition water chemistry at Peach Bottom Unit 3. This proposed change is necessary to the test since it is anticipated that main steam line radiation levels may increase by a factor of five during maximum hydrogen addition rates over the routinely experienced dose rates due to increased N-16 carry-over in the steam. The licensee has evaluated all other aspects of the proposed test under 10 CFR 50.59.

Evaluation and Discussion

We have reviewed the licensee's proposed TS changes with a focus on the capability to monitor for fuel failures and the radiological implications of the dose rate increase associated with the expected N-16 equilibrium changes during the hydrogen addition test. In addition, we reviewed the licensee's considerations of radiation protection/ALARA measures to be used during the course of the test in accordance with 10 CFR 20.1(c) and Regulatory Guide 8.8 ("Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable"). The specific details of the licensee's plans for a hydrogen inspection test were discussed via a telephone conference call on September 21, 1984. In addition, we also reviewed a description of the proposed short-term hydrogen injection test provided by the licensees's letter dated October 29, 1984, as part of its continuing program to reduce intergranular stress corrosion cracking (IGSCC) in stainless steel piping.

The primary safety function of the Main Steam Line Radiation Monitor is in Rod Drop Accident mitigation. However, the Rod Drop Accident is only a concern below 20% thermal power. The proposed hydrogen injection test will not be performed with reactor thermal power less than 20%. In addition, the capability to monitor for fuel element failures, which could result in increased occupational doses, will be maintained throughout the test by the continued capability of the Main Steam Line Radiation Monitor to detect fuel failures, the performance of routine radiation surveys, daily primary water analyses and the trends of these analyses, and the capability of downstream process monitors such as the Steam Jet Air Ejector Off-Gas monitor, to detect radioactivity from fuel failure.

The licensee has indicated that normal radiation protection/ALARA practices and procedures for the Peach Bottom site will be continued throughout the test. Additionally, main steam system dose rates will be monitored by surveys on a routine basis, particularly in accessible areas. An overall objective of the mini-test is to determine general in-plant dose rate increases as well as boundary dose rate increases, if any, as a result of hydrogen addition. Additionally, specific in-plant locations where shielding may be needed for long-term implementation of hydrogen injection will also be identified as a result of this test.

A similar test was proposed and conducted for the Dresden 2 facility following our review and approval of a similar Technical Specification change. Dose rate data taken from the Dresden test indicated that the increased main steam radiation levels could be readily accommodated by limiting access to certain turbine building areas and that shine at the site boundary meets regulatory requirements. Our review of the proposed radiation protection/ALARA measures to be implemented and the test conditions identified by the licensee leads us to the conclusion that these proposed measures and test conditions are consistent with those utilized at Dresden 2. During the May and June 1982 Dresden 2 hydrogen water chemistry test, personnel exposure problems were minimal because shielded areas were sufficiently over-shielded that the absolute increase in dose rate was very small. Access to unshielded areas was closely controlled, so that time spent in these areas was short, or if access was required, hydrogen addition was stopped temporarily to reduce main steam line N-16 activity levels. Similar precautions will be in place for the Peach Bottom 3 hydrogen water chemistry mini-test to assure no significant increase in personnel exposure.

The licensee has a radiation protection/ALARA program which has been recognized as adequate in overall NRC appraisals and includes the capability to conduct special tests and maintenance in accordance with 10 CFR Part 20 and consistent with the criteria of Regulatory Guide 8.8. An ALARA review of the test program will be performed.

Based on the adequacy of the licensee's radiation protection/ALARA program, the step-wise injection of hydrogen and the utilization of special surveys to monitor dose rate increases on site and at the site boundary, accompanied by appropriate action, including halting of the test, the capability to monitor for fuel failures, the success of the initial effort at Dresden 2 and the consistency of that effort with anticipated results, and the licensee's discussion of specific radiation protection/ALARA measures to be utilized, we find that the licensee has the capability to assure adequate worker radiological protection and keep doses as low as is reasonably achievable. Based on these capabilities and the licensee's planned actions, we conclude that the proposed Technical Specification changes are acceptable.

Environmental Consideration

This amendment involves 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 amendment involves 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 amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets 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 this amendment.

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

Dated: November 14, 1984

The following NRC personnel have contributed to this Safety Evaluation:
R. Serbu and F. Witt