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

February 12, 1993

Docket No. 50-277

Mr. George J. Beck
Manager-Licensing, MC 52A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Beck:

SUBJECT: SAFETY RELIEF VALVE ALLOWABLE OUT-OF-SERVICE TIME; PEACH BOTTOM ATOMIC POWER STATION, UNIT 2 (TAC NO. M85685)

The Commission has issued the enclosed Amendment No. 172 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application dated February 2, 1993, as supplemented by your letter dated February 8, 1993.

This amendment consists of a change to the allowable out of service time for operation of the unit with the pressure relief function of a single safety relief valve (SRV) inoperable. The change consists of a footnote to Technical Specification 3.6.D.2. The modified allowable service time described in the amendment is in effect until the next outage of sufficient duration which requires a drywell entry. The period of sufficient duration is further clarified in the enclosed safety evaluation. In no case will the amendment remain in effect beyond February 28, 1994.

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You are requested to notify the staff when you have completed the repairs to the valve and no longer require the provisions of the amendment. 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

Enclosures:
1. Amendment No. 172 to License No. DPR-44
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION w/enclosures:

Docket File	MO'Brien(2)	CGrimes, 11E21	CAnderson, RGN-I
NRC & Local PDRs	JShea	JNorberg	RJones
PDI-2 Reading	OGC	ACRS(10)	APelletier
SVarga	DHagan, 3206	OPA	
JCalvo	GHill(2), P1-22	OC/LFMB	
CMiller	Wanda Jones, P-370	EWenzinger, RGN-I	

OFC	:PDI-2/LA	:PDI-2/PM	:PDI-2/PE	:EMEB	:SRXB	:OGC	:PDI-2/D:
NAME	:MO'Brien	:JShea:rb	:APelletier	:JNorberg	:RJones	:	:CMiller:
DATE	:2/9/93	:2/9/93	:2/9/93	:2/9/93	:2/9/93	:2/9/93	:2/12/93:

with comments noted

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/S/

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NAME	:MO'Brien	:JShea:rb	:APelletier	:JNorberg	:RJones	:	:CMiller:
DATE	:2/9/93	:2/5/93	:2/9/93	:2/9/93	:2/9/93	:2/9/93	:2/12/93:

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Sincerely,

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

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   License No. DPR-44
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. George J. Beck
Philadelphia Electric Company

Peach Bottom Atomic Power Station,
Units 2 and 3

cc:

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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. 172
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 February 2, 1993, as supplemented by letter dated February 8, 1993, 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-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. 172, 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 remain in effect until the next outage of sufficient duration requiring a drywell entry. The amendment shall expire no later than February 28, 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, 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: February 12, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following page of the Appendix A Technical Specifications with the enclosed page. The revised areas are indicated by marginal lines.

Remove	Insert
147	147

LIMITING CONDITIONS FOR OPERATION

3.6.D Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, both safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.

2.
(a) From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable.*

(b) From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.

3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to atmospheric within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.D Safety and Relief Valves

1. At least one safety valve and 5 relief valves shall be checked or repleace with bench checked valves every 24 months. All valves will be tested every two cycles.

The set point of the safety valves shall be as specified in Specifications 2.2.

2. At least one of the relief valves shall be disassembled and inspected every 24 months.

3. The integrity of the relief safety valve bellows shall be continuously monitored. The switches shall be calibrated once per operating cycle. The accumulators and air piping shall be inspected for leakage using leak test fluid once per operating cycle.

4. With the reactor pressure ≥100 psig, each relief valve shall be manually opened once per operating cycle. Verification that each relief valve has opened shall either be by observation of compensating turbine bypass valve closure or load reduction or change in measured steam flow depending on the operating configuration existing during the test.

* This 30 day LCO has a one time extension until the next outage of sufficient duration that requires a drywell entry. This extension shall expire no later than Feb. 28, 1994

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-44

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. 2

DOCKET NO. 50-277

1.0 INTRODUCTION

By letter dated February 2, 1993, as supplemented by letter dated February 8, 1993, the Philadelphia Electric Company (PECo), Public Service Electric and Gas Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) submitted a request for a temporary change to the Peach Bottom Atomic Power Station, Unit 2, Technical Specifications (TS). The requested change would permit continued operation of the facility with one safety relief valve inoperable until the next outage of sufficient duration which requires a drywell entry, allowing PECo to repair the valve. The required repair to the valve must be made no later than February 28, 1994, which is the projected end of the scheduled Unit 2 mid-cycle maintenance outage. The letter dated February 8, 1993 clarified the period of time that the amendment is to remain in effect and did not change the scope of the original no significant hazards consideration. The staff corrected two typographical errors and this was discussed with the licensee in a telephone call on February 12, 1993. The licensee agreed to the change and the change did not affect the no significant hazards consideration finding.

2.0 EVALUATION

Peach Bottom is designed with 11 safety relief valves (SRVs) and 2 safety valves. These valves are designed to protect the reactor vessel by providing pressure relief during high pressure transients. The SRVs have three possible pressure relief setpoints; 1105 psig, 1115 psig, or 1125 psig. The safety valves will relieve at 1230 psig. The 11 SRVs can be manually actuated to depressurize the reactor if required. Five of the eleven SRVs (RV-2-02-71A, B, C, G, and K) are also Automatic Depressurization System (ADS) valves. Under certain transient conditions, these ADS valves will automatically open to depressurize the reactor until the low pressure Core Spray and/or Low Pressure Coolant Injection Systems can inject.

The pressure relief safety function of safety relief valve RV-2-02-71B has been declared inoperable because of a remote alarm indicating bellows failure. This valve has a relief setpoint of 1125 psig and is an ADS valve. The alarm actuates on increased pressure outside of the bellows area in the valve first stage pilot. The licensee is making a conservative call that the valve is inoperable due to a bellows failure as the valve is in a location inaccessible during power operation, precluding a determination of the actual cause of the alarm or initiation of repair work. Both Target Rock, the valve supplier, and General Electric, the plant's NSSS supplier, agree with the licensee's conclusion that the ADS and manual actuation functions of the valve are not affected by a bellows failure. The mechanical pressure relief setpoint (1125 psig) is the only valve function rendered inoperable by this failure. The NRC staff concurs with this position. In a telephone conversation on February 11, 1993, the licensee committed to verifying the remote operation capability of the valve during post failure analysis after replacement of the valve.

The licensee has performed analyses to support continued operation with one inoperable SRV confirming that the reactor can be safely depressurized even in the bounding accident conditions. The analyses specifically addressed a Main Steam Isolation Valve (MSIV) closure overpressure transient, Anticipated Transient Without Scram (ATWS) evaluation, Loss-of-Coolant-Accident (LOCA) analysis, containment integrity, and the possibility of an inadvertent opening of an SRV, including RV-2-02-71B, as presented below.

An MSIV closure overpressurization analysis was performed by the licensee using NRC-approved analysis methods. This analysis conservatively assumed that two SRVs with setpoints of 1125 psig were inoperable. The results of this analysis showed that the maximum ASME Code allowable pressure for the reactor vessel of 1375 psig (upset condition) at the bottom head would not be exceeded during such an event. This analysis indicated that a safety valve may potentially lift. This overpressurization analysis was additionally conservative in that it did not include credit for the MSIV position switch reactor trip. A subsequent plant isolation overpressurization analysis with two inoperable SRVs was performed which took credit for the functional position switch reactor trip signal. This analysis resulted in a peak reactor vessel pressure of 1225 psig, which is below the setpoint of the safety valves.

ATWS evaluations have been performed by GE to support the licensee's program (Power Rerate Program) to incorporate a future increase in rated reactor power. These evaluations concluded that reactor vessel peak pressure would be 1495 psig, which is within the ATWS peak pressure requirements (ASME emergency condition). In evaluating the ATWS analysis for the impact of having one inoperable SRV, GE qualitatively determined the power rerate condition to conservatively bound the existing condition (i.e., 100% power, one SRV inoperable) based on evaluating the relative effects of steam flow and the rod line at which the analysis was performed. This conservatism is due to the power rerate ATWS analysis inputs being based on a steam flow/heat balance for 105% power. This condition produces 776,000 lbm/hr (approximately 5.8%) more steam flow than the present rated condition. A licensee evaluation of the ATWS scenario for the present rated condition with one SRV inoperable indicates that the effective additional steam flow associated with the loss of this SRV is about 6.0%, which is approximately equal to the 5.8% additional steam flow associated with the Power Rerate ATWS condition. An additional conservatism was present in the ATWS analysis due to the analysis being performed at a more severe power/flow condition (MELLL - 121% rod line) than the present plant power/flow condition (ELLL - 108% rod line). ATWS analysis results are significantly more severe when performed at higher rod line conditions.

The licensee concluded that the Updated Final Safety Analysis Report (UFSAR) LOCA analysis is unaffected by having only ten operable SRVs. For large break LOCAs, operation of SRVs is not required. For small break LOCAs, the ADS is required to automatically depressurize the reactor vessel. As discussed above, the ADS function of this SRV is still operable.

The licensee evaluated the potential effects one inoperable SRV could have on primary containment. Primary containment design parameters were based on conditions corresponding to a Design Basis Accident (DBA) LOCA. Since SRVs would not be required to operate in a large break LOCA condition, SRV operability does not impact containment integrity.

The licensee concluded that this mode of failure would not increase the probability of an SRV inadvertently opening because a failed bellows would allow equalized pressure on each side of the first stage pilot piston which prevents the piston from moving and the valve from opening. If an SRV should inadvertently open, procedures are in place to either remedy the situation or shut down the plant. The licensee has also concluded that this mode of failure would not increase the probability of a stuck open SRV. The NRC staff agrees with the licensee that there is no increase in the probability of either the SRV inadvertently opening or sticking open since the design of the SRV is such that the functions of the pilot valve and of the ADS or manual actuations are not interdependent. A stuck open SRV is a previously analyzed accident, and the licensee has shown that adequate core cooling would be maintained with a stuck open SRV under degraded conditions.

The requested change permits operation of the facility at 100% power with one safety relief valve inoperable until the next outage of sufficient duration that requires a drywell entry in which PECo could repair the valve. This is a one-time extension which will expire after the next outage of sufficient duration requiring a drywell entry, considered by the staff to be any outage of greater than 4 days that otherwise requires a drywell entry. In addition, the one-time extension shall expire no later than February 28, 1994. This date is the scheduled completion of the next Unit 2 mid-cycle outage. TS 3.6.D.3 and the second paragraph of TS section 3.6.D.2(a), which together impose a 7-day LCO followed by a plant shutdown if two SRVs are inoperable, will remain in effect. The NRC staff, including the Reactor Systems and Mechanical Engineering branches, concurs with the licensee's analyses and evaluations which show that the design basis for the pressure relief system are still met with one inoperable SRV, and continued operation with one inoperable SRV does not place the plant in an unanalyzed condition. The NRC staff finds that this proposed one-time change to the TS for Peach Bottom Unit 2 is acceptable.

3.0 EXIGENT CIRCUMSTANCES

In the licensee's February 2, 1993 letter, they requested that their application for the license amendment be processed as involving exigent circumstances.

The Commission's regulation, 10 CFR 50.91, provides special exceptions for the issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. An exigency is a case in which the staff and the licensee need to act quickly and time does not permit the Commission to publish a Federal Register notice allowing 30 days for prior public comment, and the Commission also determines that the amendment involves no significant hazards considerations. In this instance, Peach Bottom, Unit 2, will face a TS required shutdown during a period of high demand and at a time when one of the licensee's other generating units is shutdown for a refueling outage. In accordance with 10 CFR 50.91(a)(6)(i)(B), the Commission used local media to provide reasonable notice to the public in the area surrounding the Peach Bottom Atomic Power Station facility of the licensee's amendment and of the Commission's proposed determination that a no significant hazards consideration is involved.

The NRC published a public notice of the proposed amendment, issued a proposed finding of no significant hazards consideration and requested that any comments on the proposed no significant hazards consideration be provided to the staff by the close of business on February 12, 1993. The notice was published in the York Daily Record, York Dispatch, Lancaster New Era and the Lancaster Intelligencer-Journal on February 5, 1993. The notice was also published in the Cecil Whig on February 9, 1993 and in the Bel Air Aegis on February 10, 1993.

The potential shutdown results from the TS requirement that continued reactor operation is permissible only during the 30 days following the date that the safety function of one relief valve is made or found to be inoperable, unless the valve is sooner made operable. The licensee was operating at 100% reactor power on January 18, 1993, when LCO 3.6.D.2(a) was entered due to a failed bellows alarm on safety relief valve RV-2-02-71B. The licensee attempted to troubleshoot the alarm, but was unsuccessful due to the inaccessibility of the valve during power operation. Evaluations were performed and the decision was subsequently made to request an exigent TS amendment.

The staff finds that the licensee did not deliberately or negligently cause the exigent situation to come into being. Failure of the Commission to act on the licensee's request would result in an undesirable plant shutdown.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee has analyzed the proposed amendment to determine if a significant hazard consideration exists:

The proposed amendment to the Technical Specification Section 3.6.D.2(a);

(1) does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed TS change involves a change in the allowable out of service time of an SRV. SRV's are not an initiator for any of the over pressurization transients that the SRV's are designed to help mitigate. A stuck open relief valve is an accident initiator; however, the condition which has caused the SRV to be inoperable does not increase the probability of a stuck open relief valve. The design of the valve is such that a failed bellows will equalize the pressure across the first stage pilot piston and thus prevent the piston from moving and the valve from opening.

The analysis provided in the safety discussion clearly shows that the consequences of any of the overpressurization transients do not increase with the SRV inoperable. Further, while the probability of a stuck open relief valve remains unchanged by this condition, the stuck open relief valve transient was analyzed and the station can fully handle this event.

(2) The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. The proposed TS change involves a change in the allowable out of service time for an SRV. The proposed change does not introduce any new accident initiators since there are no physical changes being made to the facility.

(3) The proposed change does not involve a significant reduction in a margin of safety. The Technical Specification Bases define two design bases for the pressure relief system; first, to meet ASME overpressurization criteria, and second, to prevent opening of the unpiped spring safety valves during normal plant isolations and load rejections. First, analysis has shown that ASME overpressurization criteria will be met. Second, this same analysis has shown that an unpiped spring safety may potentially lift if two SRVs are inoperable. However, the analysis assumptions associated with the ASME overpressurization criteria are much more conservative than the analysis assumptions required for demonstrati[ng] that unpiped spring safety valves are not opened during normal plant isolations or load rejections. Analysis of the MSIV closure overpressure event with a functional MSIV position switch scram signal and two inoperable SRVs yields a peak steam line pressure below the unpiped spring safety valve setpoint.

Based on the above considerations, including the staff's safety evaluation, the staff concludes that the amendment meets the standards set forth in 10 CFR 50.92 for a no significant hazards determination. Therefore, the staff has made a final determination that the proposed amendment involves no significant hazards consideration.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has 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. The proposed finding was issued in the local media described in Section 3.0 of this Safety Evaluation. Accordingly, the 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 the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that because the requested changes do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Pelletier
J. Shea

Date: February 12, 1993