PHILADELPHIA ELECTRIC COMPANY

NUCLEAR GROUP HEADQUARTERS

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February 8, 1993

Docket No. 50-277
License No. DPR-44

NUCLEAR SERVICES DEPARTMENT

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Peach Bottom Atomic Power Station, Unit 2
Exigent Technical Specification Change Request,
Supplement 1

Dear Sir:

Philadelphia Electric Company (PECo) hereby submits Technical Specification Change Request Number 93-05, **Supplement 1**, in accordance with 10 CFR 50.90, requesting an amendment to the Technical Specifications (Appendix A) of Operating License No. DPR-44 for Peach Bottom Unit 2. The current situation at PBAPS Unit 2 with one Safety Relief Valve (SRV) inoperable because of a failed bellows indication, conforms to the 10 CFR 50.91(a)(6) description of an exigent situation. To prevent a plant shutdown in compliance with a Limiting Condition for Operation (LCO), PECo requests that this application be processed as an exigent change as permitted by 10 CFR 50.91 (a)(6).

Currently, one of eleven SRV is inoperable. Correcting the problem requires a shutdown to access equipment in the drywell. In accordance with Technical Specification 3.6.D.2.(a), an orderly shutdown must be initiated on February 16, 1993. PECo proposes that this situation satisfies the exigent criteria set forth in 10 CFR 50.91(a)(6) because failure to act in a timely manner would result in a shutdown of the unit. PECo has applied for this amendment in a timely fashion and could not have foreseen this situation. PECo respectfully requests that the NRC process this amendment application in an exigent manner. On the morning of January 29, 1993 NRC Region I and NRC Headquarters personnel were contacted to discuss this request. **Further discussions with the NRC were held on February 5, 1993. During those discussions, the need for a supplement was recognized. This supplement, with the changes in bold face, provides the additional clarification requested on February 5, 1993.**

A001

If you have any question regarding this manner, please contact us.

Very truly yours,

G. J. Beck, Manager
Licensing Section

Enclosure: Attachments 1 and 2

cc: T. T. Martin, Administrator, Region I, USNRC
J. J. Lyash, USNRC Senior Resident Inspector, PBAPS
W. P. Dornsife, Commonwealth of Pennsylvania

Commonwealth of Pennsylvania :

:

County of Chester :

G. R. Rainey, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company; that he has read the attached Technical Specification Change Request (Number 93-05), Supplement 1, for Peach Bottom Facility Operating License DPR-44 and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

Gerald R Rainey

Vice President

Subscribed and sworn to

before me this 8th day

of February 1993

Erica A Santon

Notary Public

Notarial Seal
Erica A. Santon, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires July 10, 1995

Attachment I

Peach Bottom Atomic Power Station
Unit 2

Docket No. 50-277

License No DPR - 44

Technical Specification Change Request
93-05

"Exigent Technical Specification Change Request"

Supporting Information for Changes 5 pages

Philadelphia Electric Company (PECo), Licensee under Facility Operating License DPR-44 for Peach Bottom Atomic Power Station (PBAPS) Unit 2, requests that the Technical Specifications contained in Appendix A to the Operating License be amended. Proposed changes to the Technical Specifications are indicated by vertical bars in the margin of the affected page. The proposed revised page 147 is included in Attachment 2. **The changes to the original February 2, 1993 submittal provided by this supplement are indicated by bold face.**

Description of Changes

1) The Licensee proposes that Technical Specification 3.6.D.2.(a) be revised to include an asterisk and that page 147 of PBAPS Unit 2 Technical Specification be revised to include the asterisk statement: "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 February 28, 1994.**".

Safety Discussion

In performing the safety evaluation for this proposed change, a conservative assumption was made that the bellows of one Safety Relief Valve (SRV) has failed. In July of 1992, the "Bellows Leaking" annunciator alarmed. When the alarm was investigated, no problems with the valve or the bellows were found. The bellows were vacuumed to remove moisture and the pressure switch was calibrated. Together with the valve manufacturer (Target Rock) and General Electric (GE) we concluded that this alarm was most likely caused by moisture in the bellows/pressure switch interface. This moisture was heated during startup and the increased pressure brought up the alarm. We do not know the cause of the present alarm; however, we have conservatively declared the SRV inoperable and for the purposes of this evaluation we have assumed the bellows has failed.

At this time, SRV RV-2-02-71B has been declared inoperable for the pressure relief function because of a possible failed bellows. Incorporation of this Technical Specification change will allow reactor operation until the next outage of sufficient duration to make repairs and which requires a drywell entry. The analyses that were performed to support this change request confirm that the reactor can be safely depressurized even in the bounding accident conditions (i.e., MSIV closure overpressure transient and ATWS condition) with only ten SRVs available.

Peach Bottom is designed with eleven SRVs and two safety valves. These valves are designed to protect the reactor vessel by providing pressure relief during high pressure transients. There are three different setpoints; 1105 psig, 1115 psig, or 1125 psig, for the SRVs. The safety valves will relieve at 1230 psig.

Five of the SRVs (RV-2-02-71A, B, C, G, and K) are also Automatic Depressurization System (ADS) valves. Under certain transient conditions, these valves will automatically open to depressurize the reactor until the low pressure Core Spray and/or Low Pressure Coolant Injection systems can inject. All eleven SRVs can be manually actuated to manually depressurize the reactor if required.

RV-2-02-71B has been declared inoperable because of a possible failed bellows. Both Target Rock and GE have confirmed that this condition could incapacitate the pressure relief function of this valve. The automatic and manual depressurization modes are not affected. Therefore, even though RV-2-02-71B is an ADS valve, it is only inoperable with regard to Technical Specification 3/4.6.D. Specification 3/4.5.E for the ADS operability is not affected. The setpoint for this valve is 1125 psig.

Presently, Technical Specification 3.6.D.2(a) allows for reactor operation with one inoperable SRV for 30 days. As previously stated, an outage window is required to repair this valve. Incorporation of this TS change will allow for continued reactor operation until the next outage of sufficient duration that requires a drywell entry. Analyses were performed to determine the acceptability of continued reactor operation with only ten SRVs. Four areas of concern were addressed: MSIV closure overpressure transient, ATWS evaluation, LOCA analysis, and containment integrity. The probability of an SRV as an accident initiator is not affected by this inoperable SRV.

An MSIV closure Overpressurization Analysis was performed by PECo using NRC-approved analysis methods. This analysis (Reference 2) assumed that two SRVs with a setpoint of 1125 psig were inoperable. The results of this analysis show that the maximum ASME Code allowable pressure for the reactor of 1375 psig (upset condition) at the bottom head will not be exceeded during such an event. This analysis indicates that a safety valve may potentially lift. However, this ASME overpressure analysis is conservative in that it does not include credit for the MSIV position switch scram. A subsequent plant isolation overpressure analysis with two inoperable SRVs was performed which took credit for the functional position switch scram signal. This analysis resulted in a peak vessel pressure of 1225 psig; and therefore the unpiped safety valves will not open.

ATWS evaluations performed for Peach Bottom as part of the Power Rerate Program, concluded that reactor vessel peak pressure is 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.

The Power Rerate ATWS analysis inputs were 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. An 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 additional steam flow associated with the Power Rerate ATWS condition. Furthermore, the Power Rerate ATWS analysis was 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. Thus, it is demonstrated that an ATWS analysis for present plant conditions with one SRV inoperable would be bounded by the Power Rerate ATWS analysis, and would therefore be acceptable. References 3, 4 and 5 provide inputs to the above assessments.

The 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 previously stated, the ADS function of this valve has not been incapacitated in any way. Therefore, plant responses to design basis LOCAs are not impacted by having only ten operable SRVs.

The potential effects on primary containment were also evaluated. Primary containment design parameters were based on conditions corresponding to a DBA LOCA. Since SRVs would not be required to operate in a large break LOCA condition, SRV operability does not impact containment integrity.

For the reasons stated above, continued reactor operation until the next outage of sufficient duration requiring drywell entry is acceptable with this SRV inoperable.

This mode of failure would not increase the probability of an SRV inadvertently opening because equalized pressure on each side of the bellows will prevent the first stage pilot from moving, which in turn prevents the valve from opening. If an SRV should inadvertently open, procedures are in place to either remedy the situation or shut down the plant. A stuck open SRV is an analyzed accident, and as stated in Reference 1, PECo has analyzed and taken measures to insure that adequate core cooling would be maintained with a stuck open SRV under degraded conditions.

<u>Information Supporting a Finding of No Significant Hazards</u>

The change proposed in this application does not constitute a significant hazards consideration in that:

i) <u>The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

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 SRVs 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 over pressurization 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.

ii) <u>The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.</u>

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.

iii) <u>The proposed change does not involve a significant reduction in a margin of safety.</u>

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 valve 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 demonstration 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.

Environmental Assessment

An environmental assessment is not required for the changes proposed by this Application because the changes conform to the criteria for "actions eligible for categorical exclusion" as specified in 10 CFR 51.22(c)(9).

Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed this proposed change and have concluded that they do not involve an unreviewed safety question and are not a threat to the health and safety of the public.

REFERENCES

1. PECo letter to NRC dated 4/2/81, "Information Requested by NUREG-0737, Clarification of TMI Action Plan Requirements", Item II.K.3.16.

2. Fuel Management Section Design Record File (DRF) 9299, "Peach Bottom Overpressurization Analysis"

3. GE Letter to PECo G92-PBPR-042, dated December 21, 1992

4. OPL-3 for Peach Bottom Unit 2, Cycle 10

5. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program for PBAPS Units 2 and 3," December 1992.