PHILADELPHIA ELECTRIC COMPANY

NUCLEAR GROUP HEADQUARTERS

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July 9, 1992

Docket Nos: 50-277
50-278

License No: DPR-44
DPR-56

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Peach Bottom Atomic Power Station, Units 2 and 3, Technical Specification Change Request

Reference: 1) J.W. Shea, USNRC to G. J. Beck, PECo, Request for Additional Information (RAI), date June 11, 1992

2) G. J. Beck, PECo to USNRC, Technical Specification Change Request, date May 18, 1992.

By letter dated May 18, 1992 Philadelphia Electric Company (PECo, reference 2) requested a revision to the Peach Bottom Atomic Power Station, Units 2 and 3 Technical Specifications regarding the frequency of inspection and replacement of Main Steam Safety Valves and Relief Valves. After reviewing this submittal the NRC staff concluded that some additional information was required and issued a Request for Additional Information (RAI, reference 1). The specific requests are repeated along with our response to each request. In addition, as requested in a telephone conversation between G. J. Siefert and J. W. Shea the revised Technical Specification page 147 for both units is attached for your review.

Request 1:

In the safety discussion, PECO concluded that no time-based failure mechanism is evident from the review of the as-found Surveillance Test (ST) data since 1987. The licensee is requested to provide information on any Safety or Relief Valve (SRV) ST failures seen during that period, including magnitude and direction of failures and a comparison of the observed setpoint drift with applicable ASME Boiler and Pressure Vessel Code requirements and guidelines.

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Response

Attachment 1 lists PBAPS Main Steam Safety (SV) and Relief valves (RV) with as-found first-pop set pressures outside Technical Specification 2.2.1 tolerances occurring since 1987. This data is industry-typical.

As documented in General Electric Licensing Topical Report [illegible]-31753P, "BWROG In-Service Pressure Relief Technical [illegible]ication Revision", February 1990, SVs and RVs have [illegible]ally experienced some difficulty in meeting the ±1% [illegible] tolerance criterion following a cycle of reactor operation. The ±1% tolerance used to develop the Technical Specifications stems from the original ASME acceptance criterion [illegible] new valves or for returning valves to service. Use of the ±1% [illegible]ion as an indicator of acceptable in-service performance is [illegible]stic. ANSI/ASME has acknowledged this and has modified the in-service testing criterion from ±1% to +3% per OM-1-1981, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices." A review of Attachment 1 shows that 10 of the 12 valves lifted within the current ASME OM-1 acceptance criteria of +3%.

OM-1 was first incorporated in ASME XI Section IWV by reference in the Winter 1985 addenda. ASME B & PV Code Case N-415, "Alternative Rules for Testing Pressure Relief Devices", also allows the use of OM-1 as an alternative to the requirements in earlier editions of ASME XI Section IWV for pressure relief devices.

Request 2:

Discuss the significance of the failures described in question 1 when compared to the SRV performance assumed in the Updated Final Safety Analysis Report, specifically setpoint tolerance.

Response 2:

As part of their safety design basis (UFSAR paras. 4.4.2.1, 4.4.6) the RVs/SVs are required to limit peak vessel pressure to the ASME Upset limit of 110% of design pressure (1.1 x 1250 psig = 1375 psig). The overpressure analyses in the UFSAR (Chapters 4, 14, and Appendix K Exhibit VI) utilize a +1% tolerance on the setpoints of the RVs and SVs. Technical Specification section 2.2.1 specifies an RV/SV setpoint tolerance of ±1%. The occurrences of as-found setpoints out of Technical Specification tolerances listed in Attachment 1 would not have resulted in peak vessel pressure exceeding 1375 psig during an overpressure transient.

Occurrence number 3 will be addressed since it is the limiting case based on number of valves involved and magnitude of drift.

Three RVs had high first-pop set pressures and the two SVs had low first-pop set pressures. The as-found setpoints for the remaining 8 RVs are as follows: 1 valve with no as-found data due to excessive body-to-base joint leakage during as-found testing; 2 valves - unable to locate data; 5 valves within Technical Specification tolerances (-0.7, +0.6, +0.6, +0.2, +0.2%). The three high RV setpoints would not have resulted in peak vessel pressure exceeding 1375 psig during an overpressure transient based on the following:

a. The nuclear system pressure relief system has significant excess capacity as evidenced by the following results from UFSAR Appendix K Exhibit VI for closure of all MSIVs: *

Case	Peak Vessel Press. psig
1. High neutron flux scram, all RVs/SVs functioning	1260
2. MSIV pos. switch scram/only 2 of 13 RVs/SVs functioning	<1375
3. High neutron flux scram/only 7 of 13 RVs/SVs functioning	<1350

The Case 2 result demonstrates the significant excess capacity available for the expected direct scram. Case 1, closure of all MSIVs with high neutron flux scram, is the bounding reload overpressure analysis basis. For the case 3 variation, even if 6 of the 13 RVs/SVs were not functioning, peak vessel pressure would remain below the code allowable. The high RV setpoints in occurrence 3 are bounded by case 3, even if the 3 RVs with unknown setpoints are assumed to not function.

* Appendix K Exhibit VI analyses are based on the original RV nominal setpoints of 1080, 1090, and 1100. Use of the case 2 and 3 results for qualitative evaluation of excess pressure relief capacity is acceptable since the Chapter 4 overpressure analysis results for case 1 based on the current 1105, 1115, and 1125 RV setpoints is equivalent - case 1 = 1260 psig.

b. A correlation has been made between an increased RV set point, the delay in RV operation and the effect on the maximum vessel pressure. The RVs can open at a later time in the transient and still protect the system from overpressure. Preliminary overpressure analyses were recently performed to evaluate the effect of increased delay time on the RVs. A delay time of 0.6 seconds was used, whereas a delay time of 0.4 seconds is normally used in the reload analyses. Due to system pressure ramping at the time of RV actuation (approximately 150psi/sec), this is equivalent to a setpoint increase of between 2% and 3% on all 11 RVs. This resulted in a 19 psig increase in peak system pressure for closure of all MSIVs with high neutron flux scram. Sufficient margin (>90psi) remained to the 1375 psig code limit.

c. The low setpoints of the 2 SVs would have resulted in their opening sooner in the transient, helping to reduce the peak system pressure.

Please feel free to contact us if you have any additional questions or concerns.

Very truly yours,

G. J. Beck
Manager-Licensing

cc: T. T. Martin, Administrator, Region I, USNRC
J. J. Lyash, USNRC Senior Resident Inspector, PBAPS

COMMONWEALTH OF PENNSYLVANIA :
: ss.
COUNTY OF CHESTER :

D. R. Helwig, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company; the Applicant herein; that he has read the attached Response to a Request For Additional Information for Peach Bottom Facility Operating Licenses DPR-44 and DPR-56, 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.

D R Helwig

Vice President

Subscribed and sworn to before me this 13th day of July 1992

Erica A. Santori

Notary Public

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