Station Support Department

10CFR50.90

PECO ENERGY

PECO Energy Company
Nuclear Group Headquarters
965 Chesterbrook Boulevard
Wayne, PA 19087-5691

July 7, 1994

Docket Nos. 50-277
50-278
License Nos. 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
Response to Request for Additional Information
Regarding Power Rerate Program (RAI-4)

Dear Sir:

Attached is our response to your Request for Additional Information (RAI) dated June 8, 1994 regarding our planned implementation of the Power Rerate Program at Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The Power Rerate Program was the subject of Technical Specifications Change Request (TSCR) 93-12 which was forwarded to you by letter dated June 23, 1993.

If you have any questions, please contact us.

Very truly yours,

G. A. Hunger, Jr.

G. A. Hunger, Jr., Director
Licensing

Attachment

cc: T. T. Martin, Administrator, Region I, USNRC
W. L. Schmidt, USNRC Senior Resident Inspector, PBAPS
R. R. Janati, Commonwealth of Pennsylvania

A001

COMMONWEALTH OF PENNSYLVANIA :

: ss.

COUNTY OF CHESTER :

W. H. Smith, III, being first duly sworn, deposes and says:

That he is Vice President of PECO Energy Company; the Applicant herein; that he has read the enclosed response to the NRC Request for Additional Information, dated June 8, 1994, concerning Technical Specifications Change Request (Number 93-12) 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.

William H Smith III

Vice President

Subscribed and sworn to

before me this 7th day

of July 1994.

Erica A. Santori

Notary Public

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

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI-4)
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

Question 1

"(Section 2.5.1) The evaluation did not address the effects of the increase of the bottom head pressure and the high pressure scram setpoint on the structural and functional integrity of the control rod drive system (CRDS). Please state the basis for determining the acceptability of the CRDS regarding compliance with the design code. The information provided should include the code and edition, the code allowables, the calculated maximum stresses, deformation, and fatigue usage factor for the uprated power conditions, and assumptions used in the calculations."

Response

The Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3 Control Rod Drive (CRD) Systems were evaluated for a 1035 psig reactor dome pressure and an additional 35 psid for the vessel bottom head. The CRD mechanism's structural and functional integrity were deemed acceptable for the vessel bottom head pressure of 1070 psig.

The components of the CRD mechanism, designated as the primary boundary, have been designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III. The applicable ASME B&PV Code for the initially supplied PBAPS CRDs is the 1968 Edition including the Winter 1970 Addenda. The limiting component in the CRD mechanism is the indicator tube which has a calculated stress of 20,790 psi (allowable is 26,060). The maximum stress is due to a maximum CRD internal hydraulic pressure of 1750 psig. The analysis for cyclic operation of the CRD was conservatively evaluated in accordance with the ASME B&PV Code, Subparagraph N-415.1 (Subsubparagraph NB-3222.4(d)). All requirements of Subparagraph N-415.1 (Subsubparagraph NB-3222.4(d)) were satisfied even when considering the increased power rerate vessel bottom head pressure, thereby satisfying the peak stress intensity limits governed by fatigue. It should be noted that the CRD mechanism has been successfully tested for all operational modes at simulated reactor vessel pressures up to 1250 psig saturated conditions, which bounds the rerated high pressure scram setpoint pressure of 1101 psig. Additional analysis shows that the calculated maximum fatigue usage factor for the CRD main flange, based on Subsubparagraph NB-3222.4(d), is 0.15, which is less than the allowable limit of 1. Based on the adequate stress margins and successful testing, the deformations are reasonable, satisfactory and not a concern at power rerate conditions.

The CRD system is capable of providing 250 psig differential pressure between the Hydraulic Control Unit and the reactor vessel for control rod insert and withdraw operation. At power operation, the primary scram pressure is provided by the reactor vessel pressure. Therefore, the CRD will perform all its safety function operations at rerate conditions.

Question 2

"(Section 3.3.2) The evaluation of reactor internals did not address the code and edition used for evaluating stresses and allowables for the reactor vessel and internals. Please provide such information and list the maximum stresses, fatigue usage factor and location of highest stressed areas for both the current design and the uprated power conditions."

Response

The evaluation of the reactor internals used the design acceptance criteria of the ASME B&PV Code, Section III. The specific applicable code edition for the reactor pressure vessel, including the shroud support, is the 1965 Edition including the Winter 1965 Addenda.

The stresses or loads for 18 reactor internal components were evaluated by either confirming that the rerate load combinations are bounded by previous analyses, or scaling stresses using conservative load ratios from these analyses. In some cases, previous analyses were repeated as required to demonstrate acceptance. The structural integrity of the component is demonstrated by comparison with applicable allowable stresses. The stress results for three of the highest stress components are shown in Table 1.

For power rerate conditions, the calculated fatigue usage factor for the most limiting component is 0.997 for the feedwater nozzle thermal sleeve. Prior to rerate, the calculated fatigue usage factor was 0.886 for this component.

TABLE 1
Comparison of Maximum Stresses and Locations in PBAPS, Units 2 and 3 for Reactor Internals at Current and Power Rerate Conditions

A. Upset Conditions

Component	Maximum Stress Location[1]	Maximum Stress Comparison[2,3] Current	Rerated
Shroud Support	Support Legs	24.5 ksi (35.0 ksi)	29.5 ksi (35.0 ksi)
Steam Dryer	Bank to Support Ring Welds	NA	23.0 ksi (25.4 ksi)
Core Plate	Bottom of T-beams	NA	26.32 psi (31.6 psi)*

B. Faulted Conditions

Component	Maximum Stress Location[1]	Maximum Stress Comparison[2,3] Current	Rerated
Shroud Support	Support Legs	39.3 ksi (52.0 ksi)	39.3 ksi** (52.0 ksi)
Steam Dryer	Bank to Support Ring Welds	NA	29.0 ksi (60.8 ksi)
Core Plate	Bottom of T-beams	NA	28.0 psi (63.28 psi)*

* Core plate buckling ΔP
** Power rerate bounded by current analysis basis

Note: 1. Maximum stress locations are locations where the margin to the allowables is the smallest.

2. Allowable values are shown in parentheses.

3. Where NA is indicated, information for PBAPS was not available. PBAPS is pre-10CFR50, Appendix B station.

Question 3

"(Section 3.3.3.2) This section states, 'Elastic-plastic methods were implemented for some components; the code requirements for these methods were met'. Table 3-4 shows that the fatigue usage factor for the feedwater nozzle is nearly 1 (= 0.997) for the power uprated conditions. Please provide a detailed discussion on the analysis methodology, assumptions and compliance with the code including edition, and the code allowables used, with regards to acceptability of stress levels and fatigue considerations."

Response

The evaluation of the feedwater nozzle at rerated power conditions uses the ASME B&PV Code, Section III, Class I, Paragraphs NB-3222 and NB-3223, 1974 Edition including Summer 1976 Addenda. According to these Subsections, the structural adequacy is met if the maximum primary plus secondary stress intensity at a location on a component is less than 3 S_m of the material, where S_m is defined as the design stress intensity as defined elsewhere in the ASME B&PV Code. If the S_m limit is not met, then plastic behavior is assumed, and the simplified elastic-plastic analysis of the ASME Code, Subparagraph NB-3228.3, can be used to determine structural adequacy.

The power uprate evaluation showed that even though the S_m limit on the range of primary plus secondary stress intensity was exceeded, the criteria of Subparagraph NB-3228.3 have been met as follows:

a. The calculated range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending stresses is 29.1 ksi which is below the 3 S_m limit of 55.1 ksi.

b. The calculated fatigue usage at the power uprate conditions, taking into account the elastic-plastic factor K_e is 0.998, which is below the code allowable limit of 1.0.

c. The largest calculated cyclic range of thermal stress is 81.1 ksi. This calculation conservatively included the mechanical stress. Since the maximum calculated cyclic range of thermal stress is less than the code allowable stress of 91.5 ksi (S_n), no thermal ratchet effect will be experienced, and the material meets the thermal ratcheting requirement.

d. The maximum analysis temperature for the material is 565° F for any stress cycle for the nozzle safe end. This is below the maximum temperature in the table in Subparagraph NB-3228.3 (i.e., 700° F for carbon steel). Thus, the material temperature does not exceed the maximum temperature permitted for the material.

e. The calculated ratio of the material's minimum specified yield strength to the minimum specified ultimate strength is 0.51 which is less than the ASME Code requirement of 0.80.

The evaluation of the feedwater nozzle shows the ASME Code stress limits are met, thus the structural integrity of the feedwater nozzle is acceptable for the power rerate conditions.

Question 4

"(Section 3.11) This section states that the design adequacy evaluation results meet the requirements of ASME, Section III, Subsection NB and USAS B31.1.0 codes for the main steam and recirculation piping systems. Please explain how both ASME III and USAS codes were used for these two safety-related Class 1 systems. State the code of record, including edition, for these systems."

Response

The design adequacy evaluations utilized data from the existing as-built piping analyses for the Main Steam and Reactor Recirculation Systems. The codes of record for the analyses performed are as follows:

A. Main Steam System - USAS B31.1.0, 1967 Edition
B. Reactor Recirculation System - ASME B&PV Code, Section III, 1980 Edition including Winter 1981 Addenda

The recirculation piping replacement modifications for PBAPS. Units 2 and 3 have revised the Reactor Recirculation System code of rec... ASME B&PV Code, Section III, 1980 Edition including Winter 1981 Addenda.

The design adequacy evaluations ensure that the requirements of USAS B31.1.0, 1967 Edition were satisfied for the Main Steam System piping and the requirements of ASME B&PV Code, Section III were satisfied for the Reactor Recirculation System piping.

Question 5

"(Section 3.12) State the code and edition for the power uprate evaluation of piping and pipe supports including anchorages. List the limiting nuclear steam supply and balance of plant systems and components, with resp... to the maximum stresses and safety margin as a result of the power uprate. Provide a discussion on the acceptability of auxiliary systems to operate at the uprated power level."

Response

The code and edition used for the power rerate evaluation of all piping and piping supports is provided in the PBAPS Updated Final Safety Analysis Report (UFSAR), Appendix A. ANSI B31.1, 1967 Edition is used for all piping and piping supports except for torus attached piping. For the torus attached piping, ANSI B31.1, 1973 Edition including Summer 1973 Addenda, and ASME B&PV Code, Section III, 1977 Edition including Summer 1977 Addenda were used.

The adequate operation at the rerated conditions of the piping and piping supports in all affected plant systems, except for the Main Steam (inside containment) and Recirculation Systems is addressed in Section 3.12, "Feedwater and BOP Piping," of NEDC-32183P, "Power Rerate Safety Analysis Report For Peach Bottom 2 & 3," dated May, 1993. The limiting Balance-of-Plant (BOP) systems are the Main Steam Relief Valve Discharge, Main Steam (outside drywell) and Feedwater Systems. In most cases, calculated pre-rerate stresses were at least 10% below code allowables to accommodate rerate. The maximum stress levels and fatigue analysis results for all BOP piping were reviewed based on the bounding increases in temperature, pressure, and flow rate and are shown in Table 3-5 of NEDC-32183P. In general, rerate conditions account for about a 2-4% increase in stress levels over current conditions. In a few cases, the pre-rerate stresses were within 5% of code allowables. However, this piping was evaluated and determined to be acceptable for rerate since the stresses were still below the code allowable limits when rerate conditions were considered.

Auxiliary (i.e., BOP) systems impacted by power rerate have been evaluated and were determined to be acceptable (i.e., meet the acceptance criteria of the code of record) for operation at rerate conditions.

Question 6

"(Section 3.12.1) This section states, 'Large bore and small bore safety-related and nonsafety-related piping and supports were evaluated for acceptability at the rerated conditions and shown to be adequate as currently designed or with minor support modifications.' Please specify piping systems and pipe supports that require modifications and described the modification to be performed."

Response

Minor pipe support modifications anticipated at the time of original submittal have been reconciled, and no piping or support modifications are required for operation at rerate conditions.

Question 7

"(Table 3-5) Footnote (**) of Table 3-5 states, 'Refined fluid transient time history analysis was performed at rerate conditions to minimize predicted loading and required modifications. Values of percent changes are not meaningful due to the change in analysis methodology.' Please provide a discussion on the methodology, assumptions and results of the refined transient analysis in comparison with the design basis time history analysis in predicting piping and pipe support loads."

Response

The original design fluid transient loads were calculated manually using conservative water hammer equations. An idealized forcing function curve based on an ideal sound speed was used. During the rerate evaluation, the fluid transient load was realistically recalculated using a qualified computer program (STEHAM) which used finite differences technique to solve the governing equation and boundary conditions. The computer generated forcing function is then used as input to the NUPIPE computer program in performing the force time history pipe stress analysis.

Based on this refined analysis, stresses for both piping and supports were maintained below code allowables, and no modifications were required.