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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 48 TO FACILITY LICENSE NO. DPR-44

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNIT NO. 2

DOCKET NO. 50-277

1.0 Introduction

By letter[1] dated July 28, 1978 and supplemented by letters[2,3] dated September 5, 26 and October 4, 1978, the Philadelphia Electric Company (the licensee) requested amendment to the Technical Specifications appended to Operating License DPR-44 for Peach Bottom Atomic Power Station Unit No. 2 (PB-2). The proposed changes relate to the third refueling of PB-2, involving the replacement of 260 exposed 7x7 fuel assemblies with a like number of fresh, two water rod, retrofit 8x8 fuel assemblies designed and fabricated by the General Electric Company, together with the reconstitution and reloading of an exposed lead retrofit 8x8 assembly, previously irradiated during Cycles 2 and 3. The proposed amendment was noticed in the FEDERAL REGISTER on September 7, 1978 (43FR39869). In support of this reload application for PB-2, the licensee has submitted a supplemental reload licensing document[4] prepared by the General Electric Company (GE), proposed Technical Specification changes[1], information relating to the reconstitution of a lead test fuel assembly[2] and responses[3] to our request[5] for additional information on the reload application.

This reload (Reload 3) is the first for PB-2 to incorporate GE's retrofit 8x8R fuel design on a batch basis. Previously, for Reload 1, four lead retrofit test assemblies (LTAs) were loaded into the PB-2 core. These assemblies have operated satisfactorily for two cycles.

The description of the nuclear and mechanical design of the Reload 3 8x8R fuel and the exposed standard 8x8 fuel design used for Reloads 1 and 2 is contained in GE's generic licensing topical report for BWR reloads[6]. Reference 6 also contains a complete set of references to GE's topical

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reports which describe GE's BWR reload analysis methods for the nuclear, mechanical, thermal-hydraulic, transient and accident calculations, together with information addressing the applicability of these methods to cores containing a mixture of 7x7, 8x8 and 8x8R fuel. Portions of the plant-specific data, such as operating conditions and design parameters which are used in transient and accident calculations, have also been included in the topical report.

Our safety evaluation(7) of GE's generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel and GE's analytical methods for nuclear, thermal-hydraulic and transient and accident calculations, as applied to mixed cores containing 7x7, 8x8, and 8x8R fuel, are acceptable. Our acceptance of the nuclear and mechanical design of the standard 8x8 fuel was expressed in the staff's evaluation(8) of the information in Reference 9. References 7 and 8 are incorporated in this safety evaluation report by reference.

As part of our evaluation(7) of Reference 6 we found the cycle-independent input data for the reload transient and accident analyses for PB-2 to be acceptable. The supplementary cycle-dependent information and input data are provided in Reference 4, which follows the format and content of Appendix A of Reference 6.

As a result of the staff's generic evaluation(7) of a substantial number of safety considerations related to use of 8x8R fuel in mixed core loadings with 8x8 and 7x7 fuel, only a limited number of additional review items are included in this evaluation. These include the plant and cycle-specific input data and results presented in References 3 and 4, the LOCA-ECCS analysis results for the reload fuel design, and those items identified in Reference 7 as requiring special attention during reload reviews.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 4, 260 fresh 8x8R fuel bundles, with a bundle average enrichment of 2.84 wt/% U-235 will be loaded into the core, replacing a like number of exposed 7x7 assemblies. The remainder of the 764 fuel assembly reload core will consist of the irradiated 7x7, 8x8 and lead 8x8R fuel assemblies exposed during the first three fuel cycles. The reference core loading

for Cycle 4 will result in eighth core symmetry, which is consistent with previous cycles.

The information provided in Section 6 of Reference 4 indicates that the fuel temperature and void dependant behavior of the reconstituted core is not significantly different from previous cycles of PB-2. Additionally, scram effectiveness, as shown in Figures 2a and 2b of Reference 4, is also similar to earlier cycles. The 1.1%$\Delta k/k$ calculated shutdown margin for the reconstituted core meets the Technical Specification requirement that the core be subcritical by at least 0.38%$\Delta k/k$ in the most reactive operating state with the single most reactive control rod fully withdrawn and all other rods fully inserted. Finally, Reference 4 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by 3.3%Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria can be achieved by the Standby Liquid Control System.

2.2 Thermal-Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

As stated in Reference 7, for BWR cores which reload with GE's retrofit 8x8R fuel, the allowable minimum critical power ratio (MCPR), resulting from either core-wide or localized abnormal operational transients, is equal to 1.07. With this MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The 1.07 safety limit minimum critical power ratio (SLMCPR) proposed by the licensee for Cycle 4 represents a .01 increase from the 1.06 SLMCPR applicable during Cycle 3. The basis for the revised safety limit is addressed in Reference 6, while our generic approval of the new limit is given in Reference 7.

2.2.2 Operating Limit MCPR

Various transient events will reduce the MCPR from its normal operating value. To assure that the fuel cladding integrity safety limit MCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed by the licensee to determine which event results in the largest reduction in the minimum critical power ratio. Each of the events has been analyzed for each of the several fuel types (i.e., 7x7, 8x8, 8x8R), for exposure intervals corresponding to BOC4 to EOC4-1000Mwd/t and EOC4-1000 Mwd/t to EOC4.

The methods used for these calculations, including cycle-independent initial conditions and transient input parameters are described in Reference 6. Our acceptance of the values used and related transient analysis methods appear in Reference 7. Supplementary cycle-dependent initial conditions and transient input parameters used in the analysis appear in the table in Sections 6 and 7 of Reference 4. Our evaluation of the methods used to develop these supplementary transient input values have already been addressed and appear in Reference 7. The overall transient methodology, including cycle-independent transient analysis inputs, provides an adequately conservative basis[7] for the determination of transient ΔMCPRs. The transient events analyzed were load rejection without bypass, turbine trip without bypass, feedwater controller failure, loss of 100°F feedwater heating and control rod withdrawal error.

All of the transients, except for the load rejection without bypass (LR w/o BP), were analyzed by the licensee using the generic methods and assumptions described in the generic reload topical report. For Cycle 4, the licensee analyzed the LR w/o BP modelling a plant-unique load shedding recirculation pump trip which is currently installed at both Peach Bottom Unit No. 2 and Peach Bottom Unit No. 3. The pump trip results in a substantial reduction in the calculated transient ΔMCPR when compared to the LR w/o BP without recirculation pump trip (RPT). This stems from the substantial negative reactivity addition which occurs when the core void fraction rapidly increases as a result of the core flow coastdown.

The LR w/o BP, with the load shedding RPT included in the analysis, was not calculated to be a limiting event for any fuel type or exposure interval. However, in view of the uncertain MCPR benefits of the subject RPT feature as well as its undocumented reliability of the RPT system the licensee was requested to either (a) perform a reanalysis of the LW w/o BP without taking credit for the load shedding RPT or (b) document the high reliability of the pump trip system based on design, testing and related technical specification requirements. The licensee elected to reanalyze the event without taking credit for the RPT feature.

The reanalysis[3] showed that, depending on fuel types, the LR w/o BP is the most limiting transient during certain exposure intervals of Cycle 4.

Based on our composite review of the original[4] and revised analyses[3], for the 7x7 fuel types, the most limiting abnormal operational transient is the control rod withdrawal error regardless of cycle exposure. For the standard 8x8 fuel type, the most limiting event from BOC4 to EOC4-1000 MWd/t is the control rod withdrawal error, while from EOC4-1000 MWd/t to EOC4 the load rejection without bypass is most limiting. Finally, for the reload 8x8R and lead test assemblies the load rejection without bypass is limiting throughout Cycle 4. A summary of the most severe ΔMCPRs is as follows:

	ΔMCPR	
Fuel Type	BOC4 to EOC4-1000 Mwd/t	EOC4-1000 Mwd/t to EOC4
7x7	0.24	0.24
8x8	0.19	0.21
8x8R/LTA	0.18	0.21

Addition of the above ΔMCPRs to the 1.07 safety limit MCPR gives the required operating limit MCPR for each fuel type and exposure interval. Accordingly, based on the original and revised analyses, the licensee has proposed the following operating limit MCPRs for PB-2 during Cycle 4:

	OPERATING LIMIT MCPR	
Fuel Type	BOC4 to EOC4-1000 Mwd/t	EOC4-1000 Mwd/t to EOC4
7x7	1.31	1.31
8x8	1.26	1.28
8x8R/LTA	1.25	1.28

The licensee has also considered the effect of a possible fuel loading error on bundle CPR. An analysis of the most severe misoriented fuel loading error using GEs new methodology[10,11], which, as modified, has been approved[12] by the staff, shows that rotating a fresh 8x8R fuel bundle will not cause a violation of the 1.07 safety limit MCPR. Additionally, an analysis of the most severe mislocated fuel bundle using GE's standard analysis procedure, shows that mislocating a fresh 8x8R will not violate the MCPR safety limit. Thus, when PB-2 is operated in accordance with the above operating limit MCPRs the 1.07 SLMCPR will not be violated in the event of the most severe abnormal operational transients or fuel loading errors. This is acceptable to the staff.

2.2.3 Fuel Cladding Integrity Safety Limit LHGR

The control rod withdrawl error and fuel loading error events were also analyzed by the licensee using methods acceptable to the Staff to determine the maximum linear heat generation rates (LHGR). The results for PB-2, Cycle 4 show that the fuel type and exposure dependent safety limit LHGRs, given in Table 2-3 of Reference 6 will not be violated should these events occur.

2.3 Accident Analysis

2.3.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License, implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors". One of the requirements of the Order was that prior to any license amendment authorizing any core reloading... "the licensee shall submit a re-evaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50.46". The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation assumptions.

For Cycle 4 the licensee has reevaluated the adequacy of PB-2 ECCS performance in connection with the new reload fuel design, using methods previously approved by the staff. The results of these plant-specific analyses are given in Reference 4.

We have reviewed the information submitted by the licensee and conclude that PB-2 will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when operated in accordance with the MAPLHGR versus Average Planar Exposure values given in Section 6 of Reference 4.

2.3.2 Control Rod Drop Accident

For the worst case control rod drop accident (CRDA) during hot startup conditions, the key plant-specific nuclear characteristics are within those used in the bounding CRDA analysis given in Reference 6. Since the bounding analysis showed that the peak fuel enthalpy does not exceed the 280 cal/gm fuel enthalpy design limit, the peak fuel enthalpy associated with a CRDA from hot startup condition for PB-2 during Cycle 4 will also be within the 280 cal/gm design limit.

Because the characteristic accident analysis input parameters for the worst case CRDA starting from cold startup conditions did not satisfy all of the assumptions of the bounding analysis the licensee reanalyzed this event on a plant-specific basis. The results showed the peak fuel enthalpy to be less than the 280 cal/gm limit which is acceptable.

2.4 Overpressure Analysis

The licensee has reanalyzed the limiting pressurization transient to demonstrate that the ASME Boiler and Pressure Vessel Code requirements are met during Cycle 4. The methods used for this analysis, when modified to account for one failed safety valve, have been previously approved by the staff. The acceptance criteria for this event is that the calculated peak transient pressure not exceed 110% of design pressure, i.e., 1375 psig. The reanalysis shows that the peak pressure at the bottom of the reactor vessel is equal to 1315 psig for worst case end-of-cycle conditions, even when assuming the effects of one failed safety valve. This is acceptable to the staff.

2.5 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for PB-2, Cycle 4 using the methods described in Reference 6. The results show that the channel hydrodynamic and reactor core decay ratios at the least stable operating state, (corresponding to the intersection of the natural circulation curve and 105% rod line on the power-flow map) are below the 1.0 Ultimate Performance Limit decay ratio proposed by GE.

The staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition. This condition could be reached during an operational transient from high power if the plant were to sustain a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as reload fuel designs change. The staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capability of the analytical metnods to accurately predict decay ratios.

The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. Although a final test report has not as yet been received by the staff for review, it is expected that the test results will aid considerably in resolving the staff concerns.

For Cycle 3, the staff, as an interim measure, added a requirement to the PB-2 Technical Specifications which restricted planned operation in the natural circulation mode. Continuation of this restriction will also provide a significant increase in the reactor core stability operating margins during Cycle 4 so that the decay ratio is <1.0 in all operating modes. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of PB-2 during Cycle 4 to be acceptable.

4.0 Peach Bottom Unit No. 2 Test Programs

4.1 Lead Test Assemblies and Developmental Channels

As part of the first reload of PB-2, twelve developmental channels and four lead test assemblies (LTAs) of the retrofit 8x8 fuel type were inserted in the core for qualification irradiation testing. Examinations (13,14) of these channels and LTAs have shown that they are performing acceptably. For Cycle 4, one of the LTAs will be reconstituted by replacing two exposed fuel rods with two fresh 3.00 wt/% fuel rods, in order to permit destructive examination of the irradiated test rods. Analyses which have been performed by the licensee(2) demonstrate that the planned reconstitution will not adversely impact fuel bundle performance during normal, abnormal operational transient and postulated accident conditions. Based on our review, we approve the continued use of the four LTAs and twelve developmental channels at Peach Bottom Unit No. 2 during Cycle 4.

4.2 Physics Startup Testing

Several of the key reload safety analysis inputs and results can be assured via preoperational testing. In order to provide this assurance the licensee will perform a series of physics startup tests, which are described in Reference 15. Based on our review this program is acceptable. A written report, describing the results of the physics startup tests, will also be provided by the licensee within 90 days of startup which is also acceptable.

5.0 Technical Specification Changes

The proposed technical specification changes(1) include a revised fuel cladding integrity safety limit MCPR, a revised exposure-dependent operating limit minimum critical power ratios (MCPR) for each fuel type, addition of a MAPLHGR vs average planar exposure curve and addition of a design maximum total peaking factor for the reload 8x8R fuel assemblies.

The revised 1.07 safety limit MCPR results in a .01 increase from the 1.06 safety limit MCPR (SLMCPR) used during Cycle 3. Based on our generic review[7], we find the use of a 1.07 SLMCPR for PB-2 during Cycle 4 to be acceptable. Also, based on the discussions appearing in Section 2.2.2 herein, the staff finds the proposed operating limit MCPRs, as modified[3] to reflect the reanalysis of the load rejection without bypass transient to be consistent with and adequately supported by the Reload 3 safety analyses.

The proposed 8x8R design maximum total peaking factor of 2.51, used in connection with the APRM Flux Scram and APRM Rod Block Trip Settings has been reviewed and found to be acceptable. Finally, we find that the proposed MAPLHGR vs average planar exposure curve is adequate to assure conformance with the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 for the reload 8x8R fuel assemblies.

6.0 Environmental Considerations

We have determined that the amendment does not involve a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

7.0 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: October 16, 1978

References:

1. Philadelphia Electric Company letter (Bradley) to USNRC (Denton) dated July 28, 1978.

2. Philadelphia Electric Company letter (Bradley) to USNRC (Denton) dated September 5, 1978

3. Philadelphia Electric Company letters (Bradley) to USNRC (Denton) dated September 26. 1978 and October 4, 1978 transmitting NEDO-24132, Rev. 1.

4. "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit No. 2, Reload No. 3" NEDO-24132, July 1978.

5. USNRC letter (Ippolito) to Philadelphia Electric Company (Bauer) dated September 12, 1978.

6. "Generic Reload Fuel Application", General Electric Report, NEDE-24011-P-3, dated March 1978.

7. USNRC letter (Eisenhut) to General Electric (Gridley) dated May 12, 1978, transmitting, "Safety Evaluation for the General Electric Topical Report, 'Generic Reload Fuel Application', (NEDE-24011-P)".

8. "Status Report on the Licensing Topical Report, General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by the Division of Technical Review, Office of Nuclear Reactor Regulation, USNRC, April 1975.

9. "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel", NEDO-20360 Revision 1, Supplement 4, April 1, 1976.

10. GE letter (Engle) to NRC (Eisenhut), "Fuel Assembly Loading Error" dated June 1, 1977.

11. GE letter (Engle) to NRC (Eisenhut) dated November 30, 1977.

12. NRC letter (Eisenhut) to GE (Engle) dated May 8, 1978.

13. Philadelphia Electric Company letter (Coonay) to NRC (Lear) received July 5, 1977.

14. "Boiling Water Reactor Fuel Rod Performance Evaluation Program" NEDC-23719, October 1977.

15. Philadelphia Electric Company letter (Harkins) to NRC (Lear), dated June 12, 1977.