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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 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

The Philadelphia Electric Company (the licensee) has proposed changes to the Technical Specifications of the Peach Bottom Atomic Power Station, Unit No. 2 (Reference 1). The proposed changes relate to the replacement of 292 fuel assemblies constituting refueling of the reactor core for 5th cycle operation at power levels up to 3293 Mwt (100% power).

Specific items for which the licensee has requested approval include: (1) modification of the average power range monitor (APRM) and rod-block monitor (RBM) setpoint equations, (2) deletion of the fuel densification power spiking penalty for the 8x8 fuel, (3) deletion of the reactor vessel pressure operating limit, (4) increase in the Standby Liquid Control System (SLCS) capacity, (5) use of two control rods containing hafnium control pins, and (6) extension of exposure times on the Lead Test Assemblies (LTAs).

In support of these requests the licensee provided References 2 and 3 as part of the reload application. The licensee's proposed reload with 292 fuel assemblies consists entirely of the pressurized retrofit, P8x8R, fuel design. The remainder of the 764 fuel assemblies in the core will be of mixed fuel types irradiated during the previous cycle(s).

A large number of generic considerations related to the General Electric 7x7, LTA, 8x8, 8x8R and P8x8R fuel types and mixed cores containing these fuel types, were approved by the NRC in References 4, 5 and 6. Only the additional areas of review are discussed in this safety evaluation.

The GE topical reports, References 7 and 8, provide comprehensive summaries of GE BWR reload related issues, requirements and limitations. NEDE-24011-P (Reference 7) which was approved by Reference 5 also contains values for each plant-specific datum such as steady state and operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other various design parameters. Additional plant and cycle dependent information is provided in the reload analysis. (Reference 2), which closely follows the outline of Appendix A of NEDE-24011-P (Reference 7). The above mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

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2.0 Evaluation

2.1 Nuclear Characteristics

The reference core loading for Cycle 5 is shown in Figure 1 of Reference 2. This core loading scheme results in quarter core symmetry. Section 4 of Reference 2 provided the calculated core effective multiplication and control system worth under a cold, xenon-free condition with the strongest control rod out. The minimum shutdown margin for this condition was calculated to be 1.20% Δk/k. This exceeds the minimum Technical Specification requirement of 0.38% Δk/k for this condition.

The SLCS, with its present capability (600 ppm boron), would bring the reactor to 2.6% Δk/k subcritical. To increase the shutdown capability of the alternate shutdown system above the Technical Specification requirement of 3.0% Δk/k subcritical, the licensee has proposed in section 5 of Reference 2 to increase the SLCS concentration to 660 ppm boron. At this increased concentration the SLCS will bring the core to at least 3.7% Δk/k subcritical.

Based on the data presented in sections 4 and 5 of Reference 2, both the control rod system and the SLSC (660 ppm boron) will have acceptable shutdown capability during Cycle 5.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limits

As noted in our evaluation (Reference 5) of NEDE-24011-P, GE utilizes two transient criteria in connection with fuel performance during abnormal operational transients. These criteria, or safety limits (GDC 10, 10 CFR 50, Appendix A), are intended to protect against either overstraining or overheating of the cladding during transient events.

To preclude fuel rod failure from excessive strain during transients, GE has established a 1.0% cladding plastic strain limit. The determinable core variable used to monitor the cladding strain during reactor operations is the Linear Heat Generation Rate (LHGR) of the fuel. Maximum LHGR conditions which effect the fuel locally can occur during abnormal operational conditions such as the Rod Withdrawal Error (RWE) and the Fuel Loading Error (FLE). A more detailed discussion on this safety limit, and its applicability to Peach Bottom Unit 2, Cycle 5 operations, is provided in Section 2.5.3.

To provide assurance that the fuel rods will not overheat during reactor operations the Critical Power Ratio (CPR) is monitored. The Safety Limit Minimum Critical Power Ratio (SLMCPR) which may be allowed to result from core-wide or localized transients (or from undetected fuel loading errors) is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling and that transition boiling will not occur during steady state operation as the result of the worst possible FLE. The dependence of the operating limit MCPR on the SLMCPR is discussed in Section 2.2.2.

2.2.2. Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee to determine which event results in the largest (ΔCPR) reduction in the MCPR. These events have been analyzed for both the exposed fuel and the new reload fuel. Addition of the largest (ΔCPR) ratio to the SLMCPR establishes the operating limit MCPRs for each fuel type.

2.2.2.1 Abnormal Operational Transient Analysis Methods

As stated in Section 1.0, this reload consists entirely of the pressurized retrofit P8x8R fuel design. However, the Cycle 4 reload consisted of 260 retrofit 8x8R fuel assemblies. The only difference between the P8x8R fuel and the 8x8R fuel is the prepressurization to three atmospheres with helium in the P8x8R fuel as opposed to one atmosphere of helium in the 8x8R fuel.

Our evaluation of the transient analysis methods used for second cycle, non-equilibrium cores, of the retrofit fuel design was provided in Reference 9. In Reference 9, we concluded that the 8x8R GEXL correlation used by GE in the reload analysis for non-equilibrium cores has conservatisms which are equivalent to the 7x7 and 8x8 GEXL correlations previously approved by the NRC staff. We also concluded that as equilibrium cores are approached, the conservatism in the analysis methods associated with non-equilibrium cores will diminish. To assure that this conservatism is not substantially eroded, we require that this issue be resolved prior to any operation approaching equilibrium cores.

The subject analysis for the retrofit fuel incorporated the local R-Factor distribution which appears in Table 5-2B of Reference 10. The R-Factors shown in the table were calculated using a local peaking factor distribution applicable to the unpressurized 8x8R fuel. The use of pressurized rods will have the effect of slightly reducing fuel temperatures during power operation which will result in a small reduction in the local Doppler feedback effect on local (pinwise) power peaking. GE states (References 11, 12) that the resulting difference between unpressurized 8x8R and pressurized P8x8R local power peaking is insignificant. Moreover, higher peaking in the P8x8R assemblies would tend to reduce the flatness of intrabundle peaking. Since decreased peaking (flatter power distribution) results in more rods in boiling transition in the GETAB statistical analysis, the use of the 8x8R R-Factor distribution for P8x8R reloads is considered conservative. Thus, we find the statistical safety limit, originally derived for 8x8R reloads, to be equally acceptable for P8x8R BWR reloads.

However, the non-conservative adverse effect of high flow quality (void fraction) within the P8x8R fuel assembly channels, which results from the same reduction in fuel time constant, will still be present whenever P8x8R assemblies are in the core. Thus, the transient critical bundle power in the pressurized P8x8R fuel assemblies will be decreased relative to the unpressurized 8x8R and unpressurized 8x8 assemblies. GE sensitivity studies (Reference 12) indicate that for core-wide events the P8x8R assemblies will have a slightly larger transient ΔCPR (0.1) than the unpressurized 8x8 and retrofit unpressurized 8x8R fuel types. Thus, as a result of the reduced fuel time constant, the P8x8R assemblies will require a correspondingly higher operating limit MCPR than the 8x8R/8x8 assemblies whenever the limiting transient is a rapid pressurization transient.

Therefore, considering the above discussion, when operating MCPR limits for mixed (P8x8R, 8x8R and 8x8) reload cores are established based on rapid core-wide transient events, we find it acceptable to either: (1) perform separate GETAB transient analyses (separate operating limits) for the pressurized and unpressurized fuel assemblies, or (2) perform a single GETAB transient analysis (a single operating limit) which conservatively incorporates the fuel rod thermal characteristics of the P8x8R fuel assembly. In the reload analysis for Cycle 5 of Peach Bottom Unit 2, the licensee has selected option 1, which is acceptable.

During our review of Reference 1, it was noted that the licensee proposed changes in the Technical Specifications related to the analytical treatment of the transients. The proposed change affected the scram insertion times, specifically the Reactor Protection System (RPS) logic delay time. Staff discussions with the licensee and GE revealed that GE was using 50 msec for the RPS logic delay time in the reload analyses instead of the 100 msec which is consistent with the existing Technical Specifications. The proposed change was to bring the Technical Specifications into agreement with the reload analysis. This approach is inconsistent with 10 CFR 50.59 "changes, tests and experiments." Typically such changes should be supported by a written safety evaluation which provides the bases for the changes. The safety considerations involved are: (1) the proposed change reduces the End of Cycle (EOC) ΔCPR for the limiting transient, which sets the Operating Limit Minimum Critical Power Ratio (OLMCPR) (see Section 2.2.2) and (2) the proposed change decreases the vessel pressure for the Main Steam Isolation Valve (MSIV) overpressurization event. Therefore, the proposed change may be considered an unreviewed safety question as defined in §50.59(2)(iii).

Until such time that the 50 msec RPS logic delay time is specifically approved for use in reload analyses, the calculated ΔCPR for the transient analysis will be augmented with an additional ΔCPR of 0.03 (Reference 1). Likewise, the peak calculated pressure for the MSIV overpressurization event will be increased by 5 psi.

The licensee and the NRC staff have discussed this position and both are in agreement with these determinations. Results of the licensee's analyses which include the above adjustments are discussed in Section 2.2.2.2.

2.2.2.2 Abnormal Operational Transient Analysis Results

The transients evaluated were the generator load rejection without bypass, feedwater controller failure at maximum demand, loss of 100°F feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 6, 7 and Figure 2 of Reference 1 were assumed.

The calculated systems responses and ΔCPRs for the above listed operational transients and conditions have been analyzed by the licensee. Listed below are the limiting MCPRs for the various fuel types at the specified cycle exposure.

Transient	Limiting Exposure Time	OLMCPR (8x8)	(8x8R/LTA)	(P8x8R)
Rod Withdrawal Error**	BOC 5 to EOC 5 - 1000 Mwd/t	(1.28)	(1.28)	(*)
Fuel Loading Error***	BOC 5 to EOC5 - 1000 Mwd/t	(*)	(*)	(1.30)
Load Rejection Without Bypass ****	EOC5 - 1000 Mwd/t to EOC5	(1.31)	(1.31)	(1.33)

*Not Limiting

**Includes the effects of densification power striking (see Section 6.0)

***Includes 0.02 ΔCPR allowance (see Section 2.5.3)

****Includes 0.03 ΔCPR augmentation (see Section 2.2.2.1)

Addition of the most severe ΔCPR to the safety limit (1.07) gives the appropriate operating limit MCPR for each fuel type. This sum will assure that the safety limit MCPR is not violated.

We have determined that the operating limit MCPRs listed above are acceptable for Cycle 5 operation at Peach Bottom Unit No. 2.

2.3 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 5. The faster fuel time constant of the reload pressurized P8x8R fuel results in more (thermal) energy being deposited in the fuel channel (within the reactor coolant pressure boundary) in a shorter period of time when compared with unpressurized fuel. However, GE sensitivity studies show that this more rapid energy transfer has a negligible effect on the peak system pressure associated with pressurization type transients. Nevertheless, current GE BWR system transient methods for mixed reload cores will account for this small effect via the dominant fuel type selection procedure discussed in Reference 7. Thus, we find that the effects of fuel prepressurization are adequately accounted for in vessel overpressurization analyses. Also as specified in Reference 5, the sensitivity of peak vessel pressure to failure of one safety valve has been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the overpressure design limit (1375 psi) to allow for the failure of at least one valve.

Therefore, the limiting overpressure event as analyzed by the licensee, and adjusted in accordance with Section 2.2.2.1, is acceptable.

2.4 Thermal-Hydraulic Stability

A thermal-hydraulic stability analysis was performed for this reload using the methods described in Reference 7. The results show that the fuel type dependent channel hydrodynamic stability decay ratios and reactor core stability decay ratio at the least stable operating state (corresponding to the intersection of the natural operating state curve and the 105% rod line) are 0.29 for the 8x8R/P8x8R, 0.39 for the 8x8 and 0.85 respectively. These predicted decay ratios are all below the 1.0 Ultimate Performance decay ratio proposed by GE.

Because the pressurized fuel has a shorter thermal time constant, reactor core thermal-hydraulic stability will also be affected since it involves coupled neutronic thermal-hydraulic dynamic behavior. Sensitivity studies (Reference 13) performed with GE's licensing basis stability methods indicate that the core stability decay ratio monotonically increases with increasing fuel rod gap conductance. Thus, it is to be expected that actual core stability at the least stable operating state will decrease somewhat (increased decay ratio) during the transition from unpressurized to pressurized fuel. Additional stability studies (Reference 11) have been performed by GE more recently, utilizing their licensing basis stability code and gap conductance input from their approved GEGAP-III computer code. These studies indicate that prepressurizing 8x8R fuel to three atmosphere will cause the actual core stability decay ratio to increase by approximately 0.08 for operating BWR/2&3s and approximately 0.10 for BWR/4s. However, GE has historically utilized a constant gap conductance value of 1000 Btu/hr-ft^2-°F for licensing calculations. This conservatively bounds the gap conductance values predicted by GEGAP-III for both unpressurized and pressurized fuel designs. Moreover, GE states (Reference 11) that a significant decrease in calculated decay ratios (0.2 to 0.3) would be realized if GEGAP gap conductance values were used instead of a constant value of 1000 Btu/hr-ft^2-°F. Thus, although no change in decay ratios will be predicted on a licensing basis for core reloads with pressurized fuel compared to core reloads with unpressurized fuel, GE believes that adequate conservatisms will be retained in P8x8R core stability calculations.

We have 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. Our concerns relate to both the consequences of operating with a decay ratio of 1.0 and the capability of the analytical methods to accurately predict decay ratios. The General Electric Company is addressing these NRC staff concerns through meetings, topical reports and stability test program. It is expected that the test results and data analysis, as presented in a final test report, will aid considerably in resolving the staff concerns.

Prior to Cycle 5 operation, as an interim measure, we added a requirement to the Technical Specifications which restricted planned plant 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 5. On the basis of the foregoing, we consider the thermal-hydraulic stability during Cycle 5 to be acceptable.

2.5 Accident Analysis

Our generic evaluation of the applicability of GE's accident analysis models and methods to pressurized (P8x8R) fuel as well as our evaluation of the effects of prepressurization on previously reviewed BWR accident analysis results is contained in Reference 11. Events considered by GE included the Control Rod Drop, Fuel Loading Error, and Loss of Coolant Accidents. Based on our review (Reference 6) of the information provided by GE, we agree that the methods and results for the Control Rod Drop Accident, and Fuel Loading Error, contained in Reference 7, remain valid and acceptable for pressurized (P8x8R) fuel.

2.5.1 Emergency Core Cooling System (ECCS) Appendix K Analysis

Input data and results for the ECCS analysis have been given in References 2, 15, and 16. The information presented fulfills the requirements for such analyses outlined in Reference 5. In connection with the Loss of Coolant Accident (LOCA) we concluded that the existing approved LOCA-ECCS models and methods remain valid for P8x8R fuel prepressurized with helium to three atmospheres. In addition, based on sensitivity studies performed by GE, we also conclude that prepressurizing the fuel to three atmospheres results in lower calculated peak cladding temperature for all BWR classes.

We have reviewed the analyses and information submitted for the reload and conclude that the Peach Bottom Unit 2 plant will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when it is operated at a MCPR greater than or equal to 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the LOCA, as described in Section 2.2).

2.5.2 Control Rod Drop Accident (CRDA)

The Peach Bottom Unit 2 Scram Reactivity Function at 20°C did not satisfy the requirements for the bounding analysis described in Reference 7. Therefore, it was necessary for the licensee to perform plant and cycle specific analysis for the CRDA. The results of this analysis indicate that the CRDA peak enthalpy under cold (20°C) conditions is 207 cal/gm. Therefore both cold (20°C) and hot (286°C) values are well below the 280 cal/gm design limit approved in Reference 5. We find these results acceptable.

2.5.3 Fuel Loading Error (FLE)

The licensee has considered the effects of postulated FLEs in the reload analysis. The FLE analysis for the most severe misloadings was performed using GE's revised analysis methods (References 17 and 18), which have previously been reviewed and approved by the NRC staff (Reference 19). The results show that the worst possible FLE will not cause violation of the 1.07 safety limit MCPR. We find that these results, which include the 0.02 ΔCPR allowance required by NRC to allow for the axially varying water gap for a misoriented fuel bundle, are acceptable.

The FLE limiting Linear Heat Generation Rate (LHGR) was calculated to be 17.3 Kw/ft which includes the effects of densification power spiking as required by Reference 20. Using previously accepted methods, GE calculated exposure-dependent LHGRs, which would result in 1% cladding plastic strain for the unpressurized standard 8x8 and unpressurized 8x8R fuel types. These calculated safety limit LHGRs, which appear in Reference 7, were found to be acceptable in connection with our evaluation of the generic reload topical report. One of the principle effects of prepressurization with helium to three atmospheres is to increase the fuel-to-cladding gap conductance. Thus, for the same local LHGR, pressurized P8x8R fuel temperatures, and hence fuel thermal expansion strains, will be less than for unpressurized 8x8R fuel. Put another way, pressurized P8x8R fuel could attain a somewhat higher LHGR at which 1% cladding strain occurs. However, GE has referenced the safety limit LHGRs previously calculated for unpressurized 8x8 and unpressurized 8x8R fuel for the Peach Bottom Unit 2 reload licensing application which includes a mixture of GE fuel types in addition to the P8x8R fuel in the refueled core.

Based on comparison of the approved safety limit LHGRs related to the 1% strain criteria, which appears in Reference 7, and the calculated LHGR of 17.3 Kw/ft from the FLE analysis, the limiting LHGR calculated for the misoriented pressurized P8x8R fuel is acceptable.

3.0 Control Rods With Hafnium Pins

The licensee has proposed use of two demonstration control rods containing three solid hafnium absorber pins in each wing. The hafnium absorber pins will replace standard B_4C absorber pins. The purpose of the demonstration hafnium control rods is to obtain information on the performance of hafnium in a BWR environment.

The mechanical design of the hafnium control rod is the same as the standard B_4C control rod currently in use. However, because hafnium is heavier than B_4C, each demonstration control rod will weigh 16 pounds more than the standard B_4C control rod. The effect of this increased weight will be a slight increase in the two rod scram times and a negligible increase in the core average scram times. Therefore the ΔCPR results for all abnormal operational transients, as described in Section 2.2, remain unchanged.

The licensee's use of the hafnium control rods and proposed changes to the Technical Specifications required for their use are supported by the safety evaluation provided in Reference 3. Therefore we have concluded that the licensee has met the requirements under the provisions of 10 CFR 50.59, and that the proposed use of the hafnium control rods is acceptable.

4.0 Lead Test Assemblies (LTAs)

The LTAs to be operationally extended were first inserted into the core at the beginning of Cycle 2. The licensee has stated that the four LTAs will be inspected prior to insertion for Cycle 5 to ascertain fuel bundle integrity. The information obtained from the LTA demonstration program will be used to systematically determine the impact of fuel reliability and weigh the advantages of extended exposures relative to other uranium utilization improvement methods.

Results of the safety evaluation supporting Cycle 5 operation of the reconstituted and non-reconstituted LTA fuel were provided in Reference 2. Based on results of the evaluations and analysis, the accident and transient analyses of Cycle 5 are insignificantly affected and the operating limits of Cycle 5 are also unaffected.

Therefore, we support continuation of the LTA program during Cycle 5 operation in the Peach Bottom Unit 2 reactor.

5.0 Physics Startup Testing

The safety analysis for the upcoming cycle is based upon a specifically designed core configuration. We have assumed that, after reloading, the actual core configuration will conform to the design configuration. A startup test program can provide the assurance that the core conforms to the design. We require that a startup test program be performed and the minimum recommended tests are:

1. Visual inspection of the core using a photographic or videotape record.

2. A check of core power symmetry by checking for mismatches between symmetric detectors.

3. Withdrawal and i[illegible] of each control rod to check the criticality and [illegible].

4. Comparison of predicted and measured critical insequence rod pattern for nonvoided conditions.

The startup test program submitted by the licensee for Cycle 4 remains acceptable for Cycle 5.

The licensee will submit to the NRC a brief written report of the startup tests within 90 days of the completion of the tests as required by the Peach Bottom specifications.

6.0 Technical Specifications

The proposed Technical Specification changes (Reference 1) for Cycle 5 include revised operating limit MCPRs for each fuel type in the core and changes to specific items identified in Section 1.0.

Based on our evaluation described in Section 2.2, we find the MCPRs therein listed to be consistent with and adequately supported by the Cycle 5 reload analysis, when augmented by the adjustments described in Section 2.2.2.1.

The proposed modification of the APRM and RBM setpoint equations are consistent with GE's recommended changes appearing in Section 5.2.1.5 of Reference 7. The new factors used in the equations eliminate the need to redefine the peaking factor limit with every fuel change. Because the resulting equations are equivalent and they reduce the potential for error in redefining peaking factors from cycle to cycle, we find the proposed modifications to the setpoint equations acceptable.

Deletion of the fuel densification power spiking penalty from the Technical Specification for the 8x8 fuel types has been approved by the NRC staff in Reference 20. This approval is contingent on augmenting abnormal operational conditions which affect the fuel locally, e.g., Rod Withdrawl Error and the Fuel Loading Error by the fuel densification power spike allowance. The licensee, as shown in Section 2.2.2.2 and 2.5.3, has met this requirement. Therefore, we find the requested deletion acceptable.

The design basis overpressure transient analysis found acceptable in Section 2.3 when augmented by the 5 psi specified in Section 2.2.2.1 provides sufficient margin between the reactor vessel high pressure setpoint (1055 psi) and the overpressure design limit (1375 psi) to accommodate the most severe pressurization transient. Additional conservatism is inherent in this comparison because the trend is for the pressure increase from the transient to be much less than directly proportional to the increase in initial dome pressure (Reference 5). Therefore, deletion of the reactor pressure vessel operating limit is acceptable.

Our evaluation for increasing the SLCS capacity, use of the two hafnium control rods, and continuation of the LTA program during Cycle 5 are provided in Sections 2.1, 3.0, and 4.0 respectively.

7.0 Environmental Considerations

We have determined that the amendment does not authorize 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 §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.

8.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

References

1. Application for Amendment of Facility Operating License DPR-44 dated March 3, 1980. Supplement dated April 28, 1980.

2. Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit No. 2, Reload No. 4, NEDO-24237 A, dated February 1980.

3. General Electric Proposed Peach Bottom Atomic Power Station Unit 2 Alternate Absorber Control Blade Test Program, NEDO-24231, Revision 1, January, 1980.

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

5. Safety Evaluation of the GE Generic Reload Fuel Application (NEDE-24011-P), April 1978.

6. Letter, T. A. Ippolito (NRC) to R. Gridley (GE), dated April 16, 1979, transmitting Safety Evaluation Supplement of the GE Generic Reload Fuel Application approving use of prepressurized retrofit 8x8 fuel for BWR reloads.

7. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDO-24011-P, May 1977.

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

9. Memorandum: P. Check (NRC) to T. A. Ippolito (NRC) Review of Cooper Nuclear Station Unit 1, Reload 4, dated April 11, 1979.

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

11. General Electric letter (E. Fuller) to NRC (O. Parr), dated August 14, 1978.

12. General Electric letter (E. Fuller) to NRC (O. Parr), dated June 8, 1978.

13. "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," General Electric Report NEDO-21506, dated January 1977.

14. NRC letter (O. Parr) to General Electric (G. Sherwood) dated November 21, 1978.

15. Loss-Of-Coolant Accident Analysis Report for James A. FitzPatrick Nuclear Power Station Plant (Lead Plant), NEDO-21662, July 1977.

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16. Loss-Of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit No. 3, NEDO-24082, December 1977.

17. General Electric letter (R. Engle) to USNRC (D. Eisenhut), "Fuel Assembly Loading Error," dated June 1, 1977.

18. General Electric letter (R. Engle) to USNRC (D. Eisenhut), dated November 30, 1977.

19. USNRC letter (D. Eisenhut) to General Electric (R. Engle), dated November 30, 1977.

20. USNRC letter (D. Eisenhut) to General Electric (R. Gridely), dated June 9, 1978.

Dated: June 13, 1980