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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-56

PHILADELPHIA ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION

UNIT NO. 3

DOCKET NO. 50-278

1.0 Introduction

By letter dated December 19, 1977[1], and supplemented and amended by letters dated August 30, 1977[2], January 17[3], February 2[4], February 17[5], May 8[6], and May 11, 1978[7] Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License No. DPR-56. In the letter dated December 19, 1977,[1] Philadelphia Electric Company also submitted a reevaluation[8] of the Emergency Core Cooling System (ECCS) performance in compliance with our Order for Modification of License dated March 11, 1977. The amendment would modify the Technical Specifications for the Peach Bottom Atomic Power Station, Unit No. 3 (Peach Bottom 3), to: (1) permit operation of the facility during Cycle 3 with up to 252 improved two water rod 8x8R reload fuel bundles, designed and fabricated by the General Electric Company (GE) and having an average enrichment of 2.83 wt 1% 235U, and (2) revise the Maximum Average Planar Linear Hear Generation Rates (MAPLHGRs) as determined by the reevaluation of the ECCS performance. This licensing action was noticed in the FEDERAL REGISTER on February 2, 1978 (43 FR 4468).

2.0 Background

The Licensee's proposed reload of Peach Bottom Unit No. 3 with 252 GE 8x8R (retrofit) fuel bundles for Cycle 3 (Reload 2) represents the third application of the new General Electric two water rod fuel bundle design on a batch reload basis for an operating BWR. Hatch-1 Reload 2, previously reviewed and approved[9] by the staff, utilized 168 GE 8x8R fuel bundles of the same basic design, except for a slightly lower enrichment. Cooper Reload 3 was approved by the staff[10] and utilized 24 8x8 fuel bundles and 76 8x8R bundles. Peach Bottom-3, reload 1, also previously reviewed and approved by the staff,[11] incorporated 187 single water rod 8x8 fuel bundles as replacement for an equal number of 7x7 fuel bundles discharged from the initial core. The Reload 2 fuel design for Peach Bottom 3 represents a slight modification of GE's previous single water rod 8x8 reload fuel assembly design, currently in operation in 14 domestic BWR's. The retrofit 8x8 fuel design is essentially identical to the BWR/6 Fuel Design[12] and the Hatch Unit No. 2 initial core fuel design which has already been accepted[13] by the staff for first cycle operation.

The documentation submitted in support of the proposed reload includes: (1) the GE BWR Reload 2 licensing application for Peach Bottom 3[2] which contains the related fuel design information and a description of the plant unique reload analyses performed, including the analytical methods employed, (2) a supplemental reload licensing submittal[14], which represents the results of the plant unique safety analyses performed for the second reload (except for LOCA analysis results), (3) the Peach Bottom 3 Loss of Coolant Accident analysis results for the new and exposed fuel[6], (4) other supplemental information[3,4,5,6,7] and (5) the proposed Technical Specification changes[1].

3.1 Mechanical Design Evaluation

The licensee has considered [2,14] the adequacy of the thermal-mechanical, structural and chemical design of the reload retrofit 8x8 fuel assembly for all modes of operation of the Peach Bottom 3 plant, including the effects of steady-state and normal operating transients, abnormal operating transients and postulated accident conditions. Our evaluation of the adequacy of the fuel bundle design, as reported in the mechanical design evaluation provided by the licensee, is contained in the following subsections.

3.1.1 Fuel Mechanical Design Description

The Reload 2 assembly design for Peach Bottom 3 is a modified version of the General Electric 8x8 fuel assembly design currently in operation in 14 domestic BWR's. The Peach Bottom 3 reload fuel design is very nearly the same as that described in the BWR/6 Fuel Design and Hatch Unit No. 2 initial core fuel designs,[12] reviewed by the staff for first cycle operation[13]. For identification purposes, the Reload 2 fuel design will be referred to as the "retrofit 8x8," "two water rod 8x8," or simply "8x8R," while the older 8x8 fuel design will be referred to as the "standard 8x8," "one water rod 8x8," or simply "8x8."

For comparison purposes, fuel assembly design parameters for the two fuel types (and the 7x7 design) are given in Table 3.1 herein. Except for the second water rod and the use of natural uranium at the fuel column ends, the design features of the retrofit 8x8 fuel assemblies are the same as those found in the standard 8x8 fuel assemblies currently operating in numerous BWR's. The 8x8 assemblies have exhibited satisfactory performance to-date [15].

As seen in Table 3.1, the 8x8 fuel bundle contains 63 fuel rods and one water rod whereas the 8x8R bundle utilizes 62 fuel rods and two water rods. The two water rods in the 8x8R assembly have a slightly larger diameter than the single water rod used in the 8x8 assembly. The two larger water rods permit improved axial and local power flattening in the 8x8R fuel assembly, compared with both the 7x7 assembly and single water rod 8x8 assembly.

TABLE 3-1

COMPARISON OF FUEL ASSEMBLY DESIGN PARAMETERS

	Fuel Type		
Design Parameter	7x7	8x8	8x8R
Fueled Rods/Assembly	49	63	62
Active Fuel Length (in.)	144	144	150*
Rod-to-Rod Pitch (in.)	0.738	0.640	0.640
Water/Fuel Ratio (cold)	2.53	2.60	2.75
Cladding O.D. (in.)	0.563	0.493	0.483
Cladding Thickness (in.)	0.037	0.034	0.032
Thickness/Diameter Ratio	0.0657	0.0689	0.0662
Fuel Pellet O.D. (in.)	0.477	0.416	0.410
Pellet/Clad Diametral Gap (mils)	12	9	9
Maximum Linear Heat Generation Rate (Kw/ft)	18.5	13.4	13.4

*Includes 6 inches of natural UO_2 at bottom and top of fuel column

The water rods are capped, hollow, Zircaloy tubes, with small flow holes at the top and bottom ends, to permit controlled coolant flow within the interior of the tubes. One of the water rods axially positions the seven Zircaloy-4 fuel assembly spacer grids. The fuel column of the 8x8R fuel assembly is 6 inches longer than the 144-inch stack length associated with the 8x8 fuel assemblies used for Reload 1. Additionally, several U-235 enrichments are used within each reload fuel assembly to aid in reducing the local power peaking. Gadolinium, a burnable poison, is also used to supplement the rod-to-rod enrichment pattern in the fuel bundle. That is, selected interior fuel rods contain uniformly distributed gadolinium in the form of gadolinia-urania pellets for local power shaping early in life. Gadolinium-bearing fuel rods were first incorporated as a regular design feature of the initial core of Quad Cities Units No. 1 and 2, starting in 1971 and 1972, respectively. Moreover, since 1965, a substantial number of test and production gadolinia-urania rods have been successfully irradiated to appreciable exposures(17).

The combined effects of the additional water rod, longer fuel column, smaller fuel rod diameter, radial enrichment zoning and rods with gadolinia-bearing fuel pellets result in increased operating margins (in more of the fuel rods in the bundle) with respect to the linear power density design limit and maximum fuel temperatures.

The reload 8x8R fuel assemblies also incorporate finger springs, fastened to the lower tie plate, to control coolant flow through the lower tie plate-to-channel bypass flow path. In addition, the Peach Bottom 3 reload assemblies will have two alternate path flow holes drilled in the lower tie plate orifice nozzle.

3.1.2 Materials Properties

The retrofit 8x8 fuel assembly components are fabricated with Zircaloy-2, Zircaloy-4, Type 304 stainless steel, Inconel X and ceramic uranium dioxide and gadolinia. These materials are the same as those used for the design of the standard 8x8 and 7x7 fuel assemblies. A substantial number of reactor-years of operating experience has been accumulated with these materials under BWR core environmental conditions. This experience has shown these materials to be compatible with the BWR environment and to retain their functional capability during reactor operations during the design life of the fuel.

Reference 2 provides the materials properties used in the safety analyses associated with the mechanical design of the reload 8x8 fuel bundle. The various properties are the same as those used for the mechanical design of the standard 8x8 fuel assembly.

A 1% plastic strain limit is used as a safety limit for the Zircaloy-2 fuel rod cladding. Below this safety limit, perforation of the cladding, due to overstraining, is not expected to occur. The empirical basis for this strain limit is an estimate of the strain at which an internally pressurized tube reaches plastic instability. GE bases this limit on strain capability of irradiated Zircaloy cladding segments, from fuel rods operated in several BWR's[18]. A 1% cladding plastic strain limit historically has been specified by GE as a fuel integrity safety limit for fuel consequences associated with abnormal operational transients.

We have reviewed the basis for materials properties used in the mechanical design analyses of the retrofit 8x8 fuel assembly and find them to be acceptable.

3.1.3 Fuel Rod Thermal-Mechanical Design

The thermal-mechanical evaluations of the retrofit 8x8 fuel rods are based on a maximum steady-state operating linear heat generation rate (LHGR) of 13.4 Kw/ft. The elastic stress limits for the fuel rod mechanical design, during normal and abnormal operating reactor conditions are based on the stress categories presented in the ASME Boiler and Pressure Vessel Code, Section III. The cladding is also designed to be free-standing during the fuel design lifetime. A fatigue analysis, based on Miner's linear cumulative damage rule, was performed to assure that the cladding will not fail as a result of cumulative fatigue damage. In addition, for abnormal operational transients, a value of 1% plastic strain, discussed in Section 3.1.2, is established as the safety limit, below which damage due to cladding plastic deformation is not expected to occur. A thermal-mechanical evaluation is performed to determine the equivalent local linear heat generation rate, which is established as the fuel cladding integrity safety limit for abnormal operating transient conditions. Pellet cladding interaction, waterlogging, fretting-corrosion, hydriding and lateral deflection have also been considered in the fuel rod mechanical design. We have reviewed the information provided by the licensee in the above thermal-mechanical design areas. Our evaluation is reported herein.

Cladding Stress Analysis

The elastic stress limits for the fuel rod cladding utilize the Tresca maximum shear stress theory to calculate stress intensities, which are then compared with the stress intensity limits given in Table 2-6 of Reference 18. The maximum shear stress theory for combined stress, as well as the stress limits, were also used by GE for the design of the 7x7 and 8x8 fuel rods. The results of the cladding stress analyses, using the stress models appearing in Table 2-8 of Reference 18, show that the calculated maximum stress intensities are all well within the applicable stress intensity limits, during all normal and abnormal operating conditions. The analyses include load cycles derived from power changes such as those occurring during startup and shutdown, for both hot and cold conditions. Daily load changes and overpower conditions are also included in the stress evaluations. The stress evaluations incorporate the effects of fuel densification power spiking to substantiate the 13.4 Kw/ft design limit LHGR. On the basis of the information provided by the licensee, including actual BWR operating experience with 7x7, 8x8, and 8x8R fuel assemblies, designed to the above stress limits, the staff finds the fuel rod cladding stress intensity limits, the staff finds the fuel rod cladding stress analysis results to be acceptable.

Cladding Collapse Analysis

Cladding collapse potential has been assessed as part of the overall thermal-mechanical design evaluation of the retrofit 8x8R fuel rods. A collapse analysis was performed using the generic methods described in the SAFE-COLAPS Model[19]. This model has been previously approved[20] by the staff. The limiting collapse criteria assumes an instantaneous increase of 250 psi in the hot full power reactor core pressure[21], due to turbine trip with bypass failure. This event can occur at any time during the life of the fuel assembly. For Peach Bottom 3, the maximum pressure increase, for the most severe pressurization transient (load rejection with bypass failure) during Cycle 3, is less than 250 psi. Thus, the generic analysis is conservative. Additionally, the analysis includes the effect of fuel densification power spiking on cladding temperature. Finally, cladding collapse has never been observed in operating BWR fuel rods. The staff, therefore, finds the cladding collapse analysis results to be acceptable for Peach Bottom 3 during Cycle 3.

Fatigue Analysis

The fatigue analysis uses Miner's linear cumulative damage rule[22]. The fuel rod location GE considers subject to the greatest fatigue damage is the fuel rod clad tube-to-end plug weld juncture. The cyclic loads considered in the analysis are coolant pressure and thermal gradients as described in

Tables 2-12 and 2-13 of Reference 18. The cyclic loads are reported by GE to be representative of a four-year residence time, at maximum thermal gradients corresponding to beginning of life conditions.

The staff considers the fatigue damage limit, as described by GE, to be adequate(23). Moreover, the results of the fatigue analysis, using the stress models appearing in Table 2-8 of Reference 18, show that the cumulative fatigue damage is well within the fatigue damage limit.

Fuel Cladding Integrity Safety Limit LHGR

In order to avoid fuel rod rupture, due to excessive cladding strain caused by fuel pellet expansion, GE has established a cladding plastic diametral strain limit of 1%. Using the previously accepted methods for calculating cladding strains, exposure-dependent linear heat generation rates (LHGR's), corresponding to 1% cladding plastic diametral strain were determined by General Electric. The corresponding LHGR's for the UO_2 fuel rods are approximately 25, 23 and 20 Kw/ft at 0, 20,000 and 40,000 Mwd/t, respectively. However, because urania-gadolinia fuel material has a lower thermal conductivity and melting temperature than urania fuel, the LHGR's corresponding to 1% plastic strain for the gadolinia bearing fuel rods in the 8x8R fuel assemblies are lower than the above values.

For the urania-gadolinia fuel rods having the maximum gadolinia loading concentration, the calculated LHGR limits corresponding to 1% plastic strain are not less than 22.0, 20.5, and 17.5 Kw/ft for 0, 20,000 and 40,000 Mwd/t, respectively. The above LHGR's, for the maximum gadolinia concentration fuel rods, are thus established as the exposure-dependent fuel cladding integrity safety limit LHGR's for both 8x8 and 8x8R fuel rods. Fuel rods with peak pellet LHGR's below the safety limit LHGR are not expected to exhibit cladding failure due to overstraining, during the most severe abnormal operational transient event.

The adverse effects of fuel densification power spiking have not, however, been directly considered in the establishment of the above LHGR's. Thus, the staff requires that the maximum calculated LHGR's (for each fuel type) for the most severe transient event, be augmented by an amount equal to the densification power spike penalty before comparison with the above limits. On this basis, the above LHGR's are acceptable fuel cladding integrity safety limits for the consequences associated with abnormal operational transients such that no fuel damage is calculated to occur if the limit is not violated. Fuel densification effects and transient results (Section 3.4) are discussed further below.

Waterlogging

Another area of continuing generic review, which is addressed adequately in the Peach Bottom 3 reload submittal, is the potential and consequences of operating with waterlogged fuel rods. We have reviewed the safety aspects of waterlogging failures that could result from pellet cladding interaction (PCI). A survey of the available information, which includes: (1) test results from SPERT and NSRR in Japan and (2) observations of waterlogging failures in commercial reactors, indicates that rupture of a waterlogged fuel rod should not result in failure propagation or significant fuel assembly damage that would affect coolability of the fuel rod assembly. Thus, we agree that the evaluation of waterlogging failures, as presented in the Peach Bottom 3 submittal, is correct, and that cladding stress design limits would not be exceeded.

Fretting-Corrosion Wear

Fretting-corrosion wear, due to flow induced fuel rod vibration against the spacer contacts has been considered in the fuel assembly design. The fuel rod vibration and support characteristics of the retrofit 8x8 fuel design are very similar to the 7x7 and standard 8x8 fuel design. Moreover, the 8x8R fuel assembly will operate in the same core environment as the 7x7 and 8x8 assemblies. Fuel rod vibration experiments and years of actual reactor operating experience[25] has provided substantial confidence in the adequacy of the BWR fuel design relative to fretting-corrosion wear behavior.[26] Moreover, actual operating experience with lead 8x8R fuel assemblies has shown the fuel to perform adequately relative to fretting wear. In view of the similarity of the 8x8R fuel design to the older GE BWR fuel designs together with the operating conditions to be associated with the 8x8R reload assemblies, the staff finds that the fretting-corrosion wear potential of the reload fuel assemblies to be acceptably low.

Lateral Deflection

Fuel rod lateral deflection, or bowing, has been investigated by GE and considered in the 8x8R fuel assembly design. The deflection limits on the magnitude of fuel rod bowing are based on: (1) cladding stress limits and (2) rod-to-rod and rod-to-channel clearance limits. Thermal-hydraulic tests of standard 8x8 fuel[27] have demonstrated that a minimum clearance of .060 inches (design clearance is 0.157 inches) is sufficient to ensure a very low probability of local rod overheating caused by a critical heat flux condition. In the GE fuel assembly surveillance programs, more than 2400 peripheral fuel rods have been examined by bore-scopic techniques. GE studies[27,28,29] show: no observable gross bowing in the standard 8x8 design, (2) very low frequency of minor bowing, (3) calculated deflections within the design limit, and (4) no significant DNB problem at small rod-to-rod and rod-to-channel clearances, based on thermal-hydraulic testing. In view of the above, the staff agrees that there does not appear to be a significant safety concern relative to potential fuel rod lateral deflection, associated with the 8x8R fuel assembly design.

Pellet Cladding Interaction

Pellet cladding interaction (PCI) is addressed in the Peach Bottom 3 reload submittal. Since 1972, General Electric has made changes in the fuel assembly design and has recommended changes in the mode of reactor operation to reduce the incidence of PCI fuel failures. To minimize the potential for pellet ridging, a shorter, chamfered pellet with no dishing will be used. The 8x8R design also includes a higher annealing temperature for the Zircaloy-2 cladding, to achieve improved uniformity of mechanical properties. In addition to these design changes, General Electric continues to recommend specific operating procedures identified as Preconditioning Interim Operating Management Recommendations (PCIOMR's). Under these procedures, the reload fuel will be preconditioned for subsequent full power operation and power cycling by first being taken to full power at a slow ramp rate. On this basis and the thermal-mechanical stress and strain evaluations performed for the reload fuel rod design, the staff agrees that the 8x8R fuel rod design will exhibit adequate performance relative to PCI type fuel failure.

Fuel Densification

The Peach Bottom 3 Reload 2 submittal[1] references the GE densification analysis [27] previously approved[31] by the staff. The effects of fuel densification on the fuel rod are to increase the stored energy, increase the linear thermal output and increase the probability of local power spikes from axial gaps.

The primary effects of densification on the fuel rod mechanical design are manifested in the calculation for fuel/cladding gap conductance, cladding collapse time and fuel duty (stress and fatigue evaluations). The approved analytical model incorporates time-dependent gap closure and cladding creepdown for the calculation of gap conductance. The cladding collapse time calculation also includes the effect of local gaps on cladding temperature. Finally, cladding collapse has not been observed in BWR fuels.

More recent densification analyses submitted by GE[32] and approved by the staff[33] have addressed the effects of increased densification in gadolinia-urania fuels. The stored energy effects of increased densification in gadolinia-urania fuels are offset by the significantly lower LHGR in the gadolinia bearing fuel rods compared to the non-gadolinia bearing fuel rods in the bundle. With regard to densification power spiking effects, GE has shown that the offsetting effects of excess thermal expansion and axial heat transfer, not previously taken credit for, more than offsets the adverse spiking effects associated with gadolinia. Thus, the staff finds that fuel densification has been acceptably accounted for in the mechanical design of the retrofit 8x8 fuel assemblies. Fuel densification effects on transients and accident consequences are addressed separately in Sections 3.4 and 3.5 herein.

Fission Gas Release

The staff has questioned the validity of the fission gas release predictions in vendor thermal-mechanical performance codes, including GEGAP-III[24], at burnups in excess of 20,000 Mwd/t. By letter[34] dated January 18, 1978, the NRC requested that GE revise their fuel performance model to account for burnup enhanced gas release and submit the revised model for staff review within one year. The staff intends to request all licensees to provide a schedule for incorporating burnup enhanced fission gas release into their safety analysis. For Cycle 3, the fresh reload fuel will not achieve burnups at which fission gas release enhancement occurs. Thus, the effect of enhanced fission gas release on safety analyses is not a concern for the 8x8R fuel for Cycle 3. Our concern, relative to the exposed fuel, is being handled generically, as described above.

Operating Experience

The standard 8x8 fuel design is currently in operation in 14 BWR's and a substantial number of fuel bundles (>250) are in their third irradiation cycle. A detailed post-irradiation examination has been performed at Monticello on lead 8x8 test assemblies at the end of their first two cycles and indicates satisfactory performance[35]. Four lead test assemblies of the 8x8R fuel design began operation in Peach Bottom Unit No. 2, in March 1976. These four assemblies were extensively visually examined at the end of one cycle in mid-1977. The examination results have demonstrated that the 8x8R assemblies and channels are in excellent condition for continued operation.[26] These assemblies are presently operating satisfactorily in their second cycle of operation. In addition, one lead 8x8R assembly, containing several pressurized fuel rods, has completed its first cycle of operation at Peach Bottom Unit No. 3[36].

3.1.4 Fuel Assembly Structural Design

The reload 8x8 fuel assembly is designed to withstand the predicted thermal, pressure and mechanical interaction loadings occurring during handling, startup, normal operation and abnormal operational transients without impairment of functional capability. The fuel assembly is designed to sustain predicted loadings from an operating basis earthquake. Also, the design-analysis of the fuel assembly shows that the functional capabilities will not be exceeded as a result of a safe shutdown earthquake. The ability of the 8x8R assembly and its components to meet these capabilities is evaluated by (1) analyses based on classical methods and the ASME Boiler and Pressure Vessel Code which are compared against acceptance criteria (design ratios) and (2) testing programs.

The adequacy of the fuel assembly structure during normal operations and normal operating transients is based principally on stress limits and stress formulations which are consistent with the requirements of the ASME Boiler and Pressure Vessel Code Section III. Based on our review of the analysis results provided by the licensees and actual reactor operating experience for the 7x7, 8x8 and lead 8x8R fuel assemblies we find the 8x8R to be structurally adequate for normal operating conditions for the Peach Bottom 3 plant.

The adequacy of the fuel assembly structural design during abnormal operating transients principally relate to the fuel cladding integrity safety limit LHGR and cladding collapse potential. These are evaluated in Section 3.1.3 herein.

The question of the adequacy of the 8x8R fuel assembly structural design during handling, and combined earthquake and LOCA loading conditions is currently being reviewed generically by the staff via topical reports submitted by GE. At present, we have not identified a significant safety concern. However, for this licensing action we have reviewed the capability of the fuel assembly to withstand the control rod drop accident, pipe breaks inside and outside of containment, the fuel handling accident and the recirculation pump seizure accident. These are addressed separately in Section 3.5.

3.2 Nuclear Design Evaluation

Our evaluation of the Cycle 3 core nuclear design for Peach Bottom 3 consists of two parts. The first part consisted of a review of the adequacy of the reference neutronics methods, for the analysis of the reload retrofit 8x8 fuel assembly and the Cycle 3 mixed core configuration. The second part addressed the acceptability of the calculated fuel assembly and core nuclear characteristics, applicable to Peach Bottom 3, during the third cycle of operation.

3.2.1 Nuclear Design Methods

The staff has reviewed and evaluated the information presented(2) on the nuclear analysis methods. The basic calculational procedures used for generating neutron cross sections are part of General Electric's Lattice Physics Model.(37,38) In this model the many-group fast and resonance energy cross sections are computed by a GAM-type program. The fast groups are treated by integral collision probabilities to account for geometrical effects in fast fission. Resonance energy cross sections are calculated by using the intermediate resonance approximation, with energy and position-dependent Dancoff factors included. The thermal cross sections are computed by a THERMOS-type program. The model accounts for the spatially varying thermal spectrum throughout the fuel bundle. These calculations are performed for an extensive combination of parameters including fuel enrichment and distribution, fuel and moderator temperature, burnup, voids, void history, the presence or absence of adjacent control rods and gadolinia concentration and distribution in the fuel rods. As part of the Lattice Physics Model, three-group two-dimensional XY diffusion calculations for one or four fuel bundles are performed. In this way, local fuel rod powers can be calculated as well as single bundle or four bundle (with or without a control rod present) average cross sections. The General Electric Company has submitted two licensing topical reports(37,38) which describe in detail and verify the adequacy of the procedures outlined above. The staff has reviewed these reports and has concluded(39) that the methods satisfy its requirements for core physics methods. These methods are considered acceptable for the Cycle 3 core, incorporating the retrofit 8x8 fuel bundles.

The single or four bundle averaged neutron cross sections which are obtained from the Lattice Physics Model are used in either two or three-dimensional calculations. Two-dimensional XY calculations are usually performed in three-groups at a given axial location to obtain gross power distribution, reactivities and average three-group neutron cross sections for use in one-dimensional axial calculations. The three-dimensional diffusion calculations use one energy group and can couple neutron and thermal-hydraulic phenomena. These three-dimensional calculations are performed using 24 axial nodes and 1 radial node per fuel bundle resulting in about 14,000 to 20,000 spatial nodes. This three-dimensional calculation provides power distributions, void distributions, control rod positions, reactivities, eigenvalues, and average cross sections for use in the one-dimensional axial calculations. The three-dimensional calculations have been described and verified in two licensing topical reports[40,41] which were submitted by the General Electric Company. The staff has reviewed these reports and has reached the same conclusions[37] as those reached for the Lattice Physics Methods reports. These methods are also considered acceptable for the Peach Bottom Cycle 3 core incorporating the retrofit 8x8 fuel bundles.

The one-dimensional calculation referred to above is a space-time diffusion calculation which is coupled to a single channel thermal-hydraulic model. This axial calculation is used to generate the scram reactivity function for various core operating states. This one-dimensional space-time code has been compared by the General Electric Company with results obtained using the industry standard code, WIGLE, and shown to be conservative. Our consultant, Brookhaven National Laboratories, has performed an extensive study[42] of BWR scram reactivity behavior and has concluded that the end of cycle, all rods out configuration, represents the limiting condition for BWR scram system effectiveness. Thus, we conclude that the method and assumptions used by General Electric to obtain the scram reactivity curves are acceptable.

The Doppler, moderator void and moderator temperature reactivity coefficients are generated in a rudimentary manner from data obtained from the Lattice Physics Model. The effective delayed neutron fraction and the prompt neutron lifetime are computed using the one-dimensional space-time code. The power coefficient is obtained by appropriately combining the moderator void, Doppler, and moderator-temperature reactivity coefficients.

The General Electric Company has submitted a licensing topical report[43] describing the methods used for the generation of void and Doppler reactivity feedback for application to BWR design. The staff is currently reviewing this report. Based on our review to-date we find the methods used by General Electric to be acceptable for this application for the reasons discussed below.

Comparisons between calculated and measured local and gross power distributions have been presented by the General Electric Company in two topical reports[38,41]. Predicted local (intra-bundle) power distributions were compared to data obtained from critical experiments and from gamma scans performed on operating plants. Gross radial and axial power distributions obtained from operating plants have been compared with values predicted by the BWR Simulator code. These comparisons have yielded values for calculational uncertainties to be applied to power distributions. Comparisons have also been made of calculated values of cold, xenon-free reactivity and hot operating reactivity of a number of operating reactors as a function of cycle exposure. These comparisons have been used to establish shutdown reactivity requirements.

3.2.2 Nuclear Characteristics

Introduction

The reference core design[14] for Cycle 3 utilizes 252 fresh retrofit 8x8 reload fuel bundles with a bundle average U-235 enrichment of 2.83 wt %, to replace 252 exposed 7x7 fuel assemblies from the initial core. The Peach Bottom 3 core contains a total of 764 fuel bundles. Thus about 33% of the fuel bundles are being replaced for this reload. The loading pattern for Cycle 3 results in a symmetrical modified scatter loading of the Reload 2 assemblies within the core.

The reload 8x8R fuel assemblies have a total active fueled length of 150 inches. This compares with a 144-inch long pellet column incorporated in the design of the initial core 7x7 fuel bundles and the first reload 8x8 bundles. The top six inches and bottom six inches of the fuel column of the retrofit fuel assembly consist of fuel pellets with natural uranium enrichment. The remaining central 138 inches contain fuel pellets of 3.01 wt % U-235. This arrangement results in a bundle average enrichment of 2.83 wt % U-235 and a lattice average enrichment (used for nuclear analysis) of 3.01 wt % U-235. The reload bundles will also incorporate several fuel rods containing enriched gadolinia as a burnable poison for local power shaping early in the cycle.

The retrofit 8x8 fuel bundles incorporate two unfueled water rods symmetrically placed on either side of the lattice diagonal. This compares with a single water rod in the Reload 1 standard 8x8 fuel bundle design. Each of the two water rods is also slightly larger than the water rod used in the 8x8 bundles. In addition, the fuel rod outside diameter (and pellet diameter) has been decreased by 10 (and 6) mils. The effect of the two larger water rods together with the smaller pellet diameter has resulted in a decrease in the fuel to water ratio.

Power Distribution

The limits on power distribution for this reload are determined by specified acceptable fuel design limits (SAFDL) and by the accident and transient analyses. These limits are reflected in the Technical Specifications as limits on the linear heat generation rate (LHGR), average planar linear heat generation rate (APLHGR), critical power ratio (CPR) and the total peaking factor (TPF). The criterion used for the review of power distributions is that these limits are assured during normal operation. For Peach Bottom 3 this criterion is met by monitoring the gross radial and axial power distributions and by pre-calculating the local power distributions.

The licensee has conservatively calculated the local power distributions and local peaking factors, by the methods described in Section 3.2.1, over the range of exposure for Cycle 3 for the 7x7, 8x8, and 8x8R designs. The retrofit 8x8 fuel design will have a lower operating local peaking factor than would a fresh standard 8x8 bundle, but not necessarily as low as an exposed 8x8 bundle of the older design. The licensee has conservatively accounted for this by assuming a local peaking factor of 1.19 which bounds both 8x8 designs for the entire third cycle. The 7x7 assemblies were assumed to have a local peaking factor of 1.24. These local peaking factors are acceptable.

Gross power distributions (radial and axial) will be monitored by periodic TIP scans, which will be updated as needed between scans by means of the LPRM detectors. This basic method is unchanged by the reload and new fuel design. The reload fuel will have a different void and axial power distribution than the older designs, due to the additional liquid water contained in the two larger water rods and the natural uranium at the fuel column ends. The calculational method used to transform detector signal to flux and power, evaluated in Reference 42, tracks ^{235}U, ^{239}Pu and ^{240}Pu in six inch segments for each assembly. An iterative technique is used to obtain self-consistent axial power and void distributions. This calculational method will measure the axial power distribution in individual bundles of the new and older designs in an acceptable manner.

Although the gross, radial and axial distributions will change due to the change in void feedback (and radial self-flattening), this is not a major effect. Since incore methods are used to monitor APLHGR, LHGR, CPR, and TPF, the changes in gross distribution are not of safety significance.

Reactivity Coefficients

Limits on reactivity coefficients are set by the transient and accident analyses, stability analysis and General Design Criterion 11, which requires that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for rapid increases in reactivity. In the Peach Bottom 3 core, the coolant is nearly isothermal at power operating conditions, and the only significant independent coefficients of reactivity are the Doppler coefficient and the void coefficient. During startup, there is also a moderator temperature coefficient.

Because the Doppler coefficient is least negative for unexposed (plutonium-free) fuel, the "worst case," least negative condition for the Doppler coefficient is at BOC. Because the new 8x8 fuel design has a different water to fuel ratio, void distribution, Dancoff factor and pin self-shielding, the behavior of the Doppler coefficient as a function of void and exposure is somewhat different than that of the standard 8x8 fuel design. The overall value remains negative under all conditions, thus meeting the requirements of GDC 11. The licensee has considered these effects on the Doppler coefficients in the accident and transient analyses in an acceptable manner. This is discussed further in Sections 3.4 and 3.5 herein.

The void coefficient is the dominant reactivity feedback coefficient. It will always be strongly negative, under all conditions encountered during Cycle 3 reactor operation. The accident and transient analyses place lower as well as upper limits on the algebraic value of the void coefficient, depending on whether power increase or decrease events are being considered. The effect of the extra water rod in the 8x8R fuel design is to reduce the absolute magnitude of the void coefficient. The licensee has calculated the void coefficient for the individual fuel types, and for the entire core, as a function of exposure, and has selected a most negative bounding value for use in the transient analyses. The calculated void coefficient is increased in absolute magnitude by the application of a 1.25 design conservatism factor when used for core wide transient analyses. This is an acceptable conservative approach.

The licensee has not calculated a moderator temperature coefficient for this cycle. This coefficient can become slightly positive during certain conditions. However, this coefficient is important only during startup and shutdown, is very slowly acting, and is overshadowed by the Doppler coefficient. Because of this, and because no credit is taken for the moderator temperature coefficient in the safety analyses, the coefficient has no direct safety significance. Therefore, the staff finds it acceptable to exclude the moderator temperature coefficient from the safety analysis.

Shutdown Capability

Shutdown margin, reactivity control systems, and scram reactivity fall under General Design Criteria 20 through 29. When applied to this reload, these Criteria reduce to the following requirements:

- The control rods must be capable of rendering the core sub-critical in a cold, xenon-free condition, at any time in the cycle, with the highest worth control rod stuck out of the core.
- The shutdown margin and scram reactivity curve must be consistent with the accident and transient analyses.
- The Standby Liquid Control System (SLCS) must be capable of rendering the core subcritical, in a cold, xenon-free condition, with the control rods at their minimum position, at any time in the cycle.

The retrofit 8x8 fuel bundles incorporate the use of small amounts of gadolinia as a burnable poison. With burnable poison in the reload core, fuel reactivity initially decreases, as samarium builds in, then increases to a peak as the burnable poison burns out, then finally decreases monotonically until EOC, as fissile nuclide depletion becomes dominant. Thus, the point of maximum core reactivity is generally not at BOC, but occurs later in the cycle. This burnable poison depletion effect also causes some control cells to increase in worth, while others may decrease, thus causing the location of the strongest rod to change. The licensee has calculated the effective multiplication factor in a core configuration with the strongest control rod out, under a cold, xenon-free condition. This calculation gives shutdown margin directly. The calculations were done for various exposures during the cycle, and a search for the strongest control rod was done at each exposure. To ensure conservatism, the minimum expected exposure for the previous cycle was assumed in the depletion calculations. The information presented in Reference 14 shows for Cycle 3, the minimum shutdown margin is 1.1% ΔK. Therefore, the shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.38% ΔK subcritical.

The licensee has calculated the scram reactivity versus time (scram curve) for EOC conditions. This scram curve, with a 0.80 multiplier for model uncertainty and error allowance, is used in the transient analyses. Rod insertion times assumed in the calculation were the slowest allowed by the Technical Specifications. These conditions are conservative for earlier exposures, because of the decrease in rod density as the EOC condition is approached. That is, at EOC, there are fewer partially inserted rods which insert reactivity more quickly than the fully withdrawn rods. The power distribution used in the calculation at each exposure is based upon the Haling axial power and exposure distributions. Since actual EOC power distributions are generally more bottom-peaked than the Haling distribution prediction, this calculation is considered conservative. Therefore, the calculation of the scram curve is acceptable, and the second of the three requirements is satisfied.

The licensee has calculated the multiplication factor and shutdown margin for a 600 ppm solium penetaborate concentration in the coolant corresponding to the Technical Specifications basis for the SLCS. Calculations were for an exposure corresponding to the maximum fuel reactivity, with the core in a cold, xenon-free state. Additionally, control rods were assumed to be out of the core. The results show the SLCS capable of rendering the core sub-critical by at least .042 ΔK for these conditions. The third requirement is satisfied for the alternate shutdown system. Thus General Design Criteria 20 through 29 are satisfied.

Control Rod Patterns and Reactivity Worths

The limits on control rod worth originate in the accident and transient analyses. Additionally, it is required that reactivity additions resulting from a single control rod notch should not result in a period which the operator cannot safely control. However, the maximum worth of one notch has never been excessive in the power range for an operating BWR.

The rod drop accident requires limits on dropped rod worth during startup, and the inadvertent rod withdrawal transient requires limits on individual rod worth during power operation. During startup, the maximum dropped rod worth is limited by limiting the possible control rod withdrawal patterns. The patterns are enforced by the Rod Sequence Control System (RSCS). This pattern restriction is independent of fuel type, and remains acceptable.

During power operation, the voided condition of the moderator greatly reduces the worth of a dropped rod, and the rod drop accident consequences are not limiting. Therefore, above 30% power, the RSCS is automatically disengaged and there is no safety-related system to control rod patterns. Further discussion may be found in the evaluation of the Control Rod Drop Accident appearing in Section 3.5.4 herein.

The limits on rod worth resulting from the analysis of the rod withdrawal error transient are enforced by means of the Rod Block Monitor System (RBM). When a control rod is selected for movement, the nearest LPRM detectors are automatically monitored, and a rod block is effected when the local power increase reaches the RBM setpoint. Thus, the RBM restricts the control rod worth through the local power coefficient, rather than via control rod patterns. This system is also independent of fuel type, and remains acceptable. Further discussion is provided in the evaluation of the Rod Withdrawal Error transient in Section 3.4 herein.

Reactivity of Fuel in Storage

The Technical Specification requirement for the storage of fuel in the Peach Bottom 3 spent fuel storage pool is that the effective multiplication factor, K_{eff}, of the fuel, as stored in the fuel storage racks, is less than 0.90 for normal storage conditions. This requirement is met if the uncontrolled infinite lattice multiplication factor, K_∞ , of a 8x8R fuel bundle in the reactor core configuration is less than or equal to 1.30. The 8DRB283 reload fuel bundle, at the peak reactivity point, at 65°C, has a maximum K_∞ of 1.198 for the enriched UO_2 fuel zone and 0.8810 for the naturally enriched UO_2 at the ends of the fuel column. Thus, the reload fuel meets the Technical Specification fuel storage subcriticality requirements.

3.3 Thermal and Hydraulic Design Evaluation

Our review of the thermal-hydraulic design for Cycle 3 of Peach Bottom 3 consisted of two parts. The first part addressed the applicability of the referenced and described thermal and hydraulic models and methods[2], for the analysis of the Cycle 3 core. Since the reconstituted core incorporates three different fuel bundle types, i.e., 7x7, 8x8 and 8x8R, the applicability of the thermal and hydraulic methods to the new retrofit 8x8 fuel bundle design was reviewed, along with a review of the adequacy of the methods for mixed cores. The second part consisted of a review of the thermal-hydraulic analysis results. The results for Peach Bottom 3 included the statistical determination of a new fuel cladding integrity safety limit MCPR for the reconstituted core and the channel and reactor core stability decay ratios.

3.3.1 Steady-State Hydraulic Methods

The core steady-state hydraulic analysis is performed to establish flow, pressure, enthalpy, void, and quality distributions within the core. This analysis also establishes initial reactor coolant conditions for reactor physics calculations and the analysis of anticipated operational transients. The hydraulic model of the reactor core includes descriptions of the orifices, lower tie plates, fuel rods, fuel assembly spacers, upper tie plates, fuel channels and core bypass flow paths. The core steady-state hydraulic model is composed of separate effects models, which simulate various pressure loss characteristics, and composite models, which simulate the channel-by-channel and core bypass flow paths.

The separate effects hydraulic models of the core and channel components consider frictional, local, elevation, and acceleration hydraulic pressure loss characteristics. The frictional characteristics of the core components are modeled by use of the single phase frictional pressure drop equation with a two-phase friction multiplier. The use of this equation and multiplier requires correlations for the friction factor, f, and two-phase multiplier, ϕ^2_{TPF}. GE has correlated these multipliers, on a best-estimate basis, to a significant amount of multi-rod geometry data[45], that are representative of modern BWR fuel bundles. The largest collection of these data was acquired from the ATLAS loop during development testing for the GEXL correlation. The data for these correlations cover the range of BWR conditions. On this basis, the use of these correlations is appropriate.

The local pressure drop characteristics have been established in a manner similar to the formulation used for the frictional pressure drop characteristics. It differs to the extent that a local pressure loss coefficient is substituted for the product of friction factor and characteristic length-to-diameter ratio. This is a common hydraulic analysis procedure. This modeling has also been verified[46] experimentally throughout the range of conditions by the ATLAS loop tests for the GEXL correlation[45]. This modeling technique is used to simulate the pressure losses of the orifice, lower tie plate, spacer, upper tie plate, and lower tie plate bypass flow holes.

The acceleration pressure drop has two components, i.e., area change and density variation. The area change is modeled similar to the local pressure drop. Since an area change is generally treated in this manner, this modeling approach is acceptable. The density variation uses the same formulation as the elevation pressure drop characteristic, except that it accounts for density variations along the fluid channel. This is also a standard hydraulic analysis practice, and is acceptable.

These separate effects hydraulic characteristics are utilized to simulate the hydraulic conditions through the orifices, lower tie plates, fuel rods, water rods, fuel rod spacers, upper tie plate and fuel channel. The core bypass flow paths have been modeled from experimental[47] results and verified by analytical techniques. These tests were previously reviewed and were found to be acceptable for this use.[48]

The above separate effects hydraulic models, which simulate reactor core component pressure losses and flow paths, permit a composite model of a single fuel channel to be simulated. The fuel channel is then categorized into a fuel "channel type." In order to reduce the number of nodes in the analysis, the fuel channels are grouped by "channel type" and modeled as a single typical channel of that type. Thus, the flow distribution of a particular fuel channel is assumed to be the same as the typical channel for that fuel channel type.

A channel type is classified by five characteristics: (1) orificing type (central or peripheral), (2) fuel geometry (7x7, 8x8, or 8x8R), (3) relative bundle power (high power or average), (4) lower tie plate type (drilled or undrilled), and (5) bypass type (finger springs or no finger springs).

With regard to the core relative bundle power distribution, sensitivity studies show[46] that classification by high power and average power density channels adequately models the core flow distribution. This is due to the fact that average channel characteristics are dominant in establishing the core pressure drop. Therefore, categorization as a function of channel power density need not be broken down into additional sub-channels. The other characteristics completely cover the range of channel type possibilities.

In order to perform channel type categorizations, each fuel channel must have the same pressure drop across its length. This is a major assumption of the steady-state hydraulic analysis. This has been shown to be valid by flow distribution and pressure drop measurements in several operating BWR's[49,50,51]. These tests further show that the pressure drop across any fuel channel or bypass flow path in the core is the same as for any other fuel channel or bypass flow path in the core. The above referenced documents have been previously accepted[49] for justification of this assumption.

The steady-state hydraulic analysis uses a digital computer code to calculate the hydraulic characteristics of the core. The code utilizes a trial and error iteration for flow rate, pressure drop, enthalpy, quality, and void distribution for each channel type. It equates the total plenum-to-plenum differential pressure across each flow path, and matches the sum of the flows to the total core flow. Comparison[46] of analytical predictions to tests performed in the ATLAS test facility as a function of pressure drop, mass flux, and bundle power show reasonably good agreement, i.e., <6% error for the range of interest. This qualifies the calculational technique and modeling for the steady-state hydraulic analysis methods for reactor pressure >800 psia and core flows >10%.

3.3.2.1 <u>Fuel Cladding Integrity Safety Limit MCPR</u>

General Design Criterion 10 requires that the reactor core be designed with appropriate margin, to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of abnormal operational transients. In order to avoid fuel damage caused by overheating of the cladding, transients are limited such that more than 99.9% of the fuel rods would be expected to avoid boiling transition during a transient event. This design basis has been previously accepted[53] for initial and reload core applications in connection with the staff's review of the General Electric Thermal Analysis Basis (GETAB) method[54]. This design basis can also be stated in terms of a statistically determined Minimum Critical Power Ratio (MCPR) safety limit. The GETAB statistical analysis procedure, including codes, correlations and analytical procedures has also been previously reviewed and approved by the staff in connection with MCPR safety limits established for initial cores and reload core applications. Our review, therefore, centered upon evaluation of the adequacy of the described statistical analysis procedures for the Peach Bottom 3 reload core, which contains three fuel types (7x7, 8x8, and 8x8R) as well as a review of the key inputs to the statistical analysis.

The nominal values of the plant process variables (e.g., core flow, dome pressure) used in the GETAB statistical analysis, are shown in Table D-2 of Reference 14. The values shown in the table correspond to the same previously approved generic 251/764 core selected for the GETAB statistical analysis, for operating BWR's which have reloaded with the standard 8x8 fuel assemblies. Substitution of the retrofit 8x8 reload fuel assemblies in the statistical analysis does not alter our previous conclusion on the acceptability of the generic core process variable parameter values selected.

The generic uncertainties associated with the core process variables, fuel bundle power determination, CHF correlation and fuel assembly manufacturing tolerances, used in the statistical analysis, appear in Table D-1 of Reference 3. The uncertainties are the same or more conservative than those shown in the GETAB report[55]. The only uncertainties in the table which are potentially reload or fuel-dependent are for TIP Readings, R-Factor, GEXL Correlation and Channel Flow Area uncertainties. The standard deviation for the TIP Readings uncertainty is 8.7% whereas the GETAB report uses a 6.3% uncertainty. The latter uncertainty is appropriate for an initial core. The uncertainty increase in TIP uncertainty to 8.7% is a consequence of the increase in the uncertainty in the bundle power measurement of a reload (exposed) core. This uncertainty is also considered to be adequate for the Peach Bottom retrofit 8x8 fuel assemblies. Table D-1 gives an R-Factor uncertainty of 1.6%, which is the same as that used for 8x8 reloads. The R-Factor uncertainty is derived from the uncertainty in the local power peaking distribution calculation. The addition of a second water rod in the retrofit fuel design is not expected to increase the uncertainty in the power distribution calculation, based on the approved neutronics methods. The 3.0% Channel Flow Area uncertainty, shown in the table, accounts for manufacturing and operationally induced variations in the free flow area within the assembly. Although the effective channel flow area for the 8x8R assembly is slightly different than for the 8x8 assembly, the manufacturing tolerances are the same. Thus, a channel flow area uncertainty of 3.0% (which is the same as the 8x8 assembly) is acceptable.

A value of 1.038 was selected for the nominal value for the retrofit 8x8 R-Factor. This compares with 1.08 and 1.095 for the 7x7 and 8x8 assembly R-Factors, used in connection with the first Peach Bottom reload. Reload 1 utilized the single water rod 8x8 fuel design. The core wide bundle histogram, used in the new GETAB statistical analysis for this reload appears in Figure D-1 of Reference 14. The CPR histogram is different from the histogram previously used in the statistical analyses of BWR 2/3/4 D-Lattice 8x8 reload cores. The new histogram indicates fewer bundles at and near the MCPR safety limit. The licensee was requested to provide additional justification to support the new retrofit 8x8 R-Factor and CPR histogram which were used in the analysis.

The additional information[5] submitted by the licensee states that the lower bundle R-Factor results from the flatter local power distribution of the 8x8R fuel design. A flatter power distribution also gives rise to a more adverse rod-by-rod critical heat flux (CHF) probability distribution and thus is more conservative relative to the number of rods calculated to be in boiling transition when the hot bundle is placed on the thermal MCPR (safety) limit. The CPR histogram used in the calculation corresponds to an all 8x8R (equilibrium cycle) reload 251/764 core. This yields the flattest bundle CPR histogram compared to non-equilibrium cycles. This also results in an adverse CHF accounting when compared to the actual or expected CPR histogram for the Peach Bottom 3 core during Cycle 3. The staff concludes that the R-Factor and bundle CPR distribution selected for the GETAB statistical analysis are appropriately conservative for Cycle 3.

The derived MCPR safety limit for Cycle 3, using the approved GETAB statistical methods and the inputs discussed above, is 1.07. This is an increase of .01 from the 1.06 safety limit applicable during Cycle 2. On the basis of the evaluation above, the staff finds the calculated 1.07 safety limit MCPR to be acceptable for Peach Bottom 3 during Cycle 3.

3.3.2.2 Thermal-Hydraulic Stability

A Cycle 3 thermal-hydraulic stability analysis, using the analytical methods discussed in Reference 2, was presented by the licensee for Peach Bottom 3. The results show that the 7x7, 8x8 and 8x8R channel hydrodynamic stability decay ratios at the least stable reactor operating state are substantially below the Ultimate Performance Limit decay ratio of 1.0 proposed by GE in Reference 2. The least stable reactor operating state corresponds to the intercept point of the 105% rod line and the natural circulation curve appearing in the plant's power flow map. The licensee has also submitted the results of the Cycle 3 reactor core thermal-hydraulic stability analysis for the least stable operating state. The results of this analysis show that the reactor core stability decay ratio is also well within the Ultimate Performance Limit decay ratio of 1.0 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 methods to accurately predict decay ratios.

The General Electric Company is addressing these staff concerns through meetings, topical reports and a stability test program. The participants of the on-going stability test program include GE and the licensee of a large BWR/4. Although a final test report has not as yet been received by the staff, it is expected that the test results will aid considerably in resolving the staff concerns.

For Cycle 2, the staff as an interim measure, imposed a requirement on Peach Bottom 3 which restricted planned operation in the natural circulation mode. This restriction which will continue during Cycle 3 will also provide a significant increase in the reactor core stability margins. On the basis of the foregoing, the staff considers the thermal-hydraulic stability of Peach Bottom 3 during Cycle 3 to be acceptable.

3.4 Abnormal Operational Transients Evaluation

Abnormal operational transients are plant system conditions, caused by a single operator error or a single equipment malfunction, which are expected to occur one or more times during the life of the nuclear plant unit. Safety (SAFDL) limits applicable for transients include the fuel cladding integrity safety limit MCPR for the 8x8R reload core and the fuel cladding integrity 1% plastic cladding strain (LHGR) safety limit for the fresh and exposed fuel designs and the reactor coolant pressure boundary (RCPB) pressure safety limit.

Our evaluation of a 1.07 safety limit MCPR for Peach Bottom 3 Cycle 3 is provided in Section 3.3.2.1. The LHGR safety limit for the retrofit 8x8, standard 8x8 and 7x7 fuel rod types is evaluated in Section 3.1.3, herein. With regard to the RCPB pressure safety limit, the maximum reactor coolant pressure achieved during the most severe abnormal operational transient is bounded by the limiting overpressurization event (MSIV closure with an indirect high flux scram) evaluated in Section 3.6 herein. The reactor vessel pressure safety limit applicable to Peach Bottom 3 during abnormal operational transients is that permitted by the ASME Boiler and Pressure Vessel Code Section III, Class 1, which permits pressure transients up to 10% above the reactor vessel design pressure. Since the design pressure of the RCPB is 1250 psig, the pressure safety limit for both abnormal operational transients and the limiting overpressurization event is 1375 psig.

We have reviewed both the methods used for simulating the fuel, core and plant system performance during transients, along with the acceptability of the calculated transient results relative to the above safety limits. Our review of the transient methods was limited to an evaluation of the applicability and adequacy of the described and referenced transient codes, correlations and analytical procedures for the Peach Bottom 3, Cycle 3 core and the new retrofit 8x8 fuel design. Our evaluation of the transient methods is reported in Section 3.4.1. The results of the transient analyses for Cycle 3 of Peach Bottom 3 are evaluated in Section 3.4.2.

3.4.1 Transient Analysis Methods

3.4.1.1 Transient Analysis Methods for Local Events - Rod Withdrawal Error

The control rod withdrawal error is an abnormal operational transient which affects only a limited number of fuel assemblies in the core. The radial peaking factor increases substantially in the fuel assemblies in the immediate vicinity of the withdrawn control rod. Thus, this transient is of safety concern with regard to potential fuel rod overheating (MCPR) and clad overstraining (1% plastic strain). Since the rate and magnitude of the gross core power increase from this event is low, the reactor pressure increase is not large enough to be of concern relative to the RCPB pressure safety limit.

The method used to calculate the consequences of this transient involves a series of steady-state calculations. The simulated core is assumed to be at its most reactive exposure, with no xenon or samarium present. The rod pattern is chosen such that the maximum worth control rod is fully inserted and the laterally adjacent or diagonally adjacent bundles are at their thermal operating limits. A series of steady-state calculations is then performed for succeeding positions of the worst case control rod using the BWR Simulator Code which calculates the response actions of the Rod Block Monitor (assuming the most adverse detector failure allowed by the Technical Specifications). The results are then used to select a setpoint for the Rod Block Monitor such that the two fuel integrity safety limits are not violated.

This procedure of using a series of steady-state calculations to approximate the transients' behavior is the standard analysis method for all GE BWR reloads. Past analyses and reviews have shown that, even at the maximum control rod drive withdrawal speed and rod worth, the rate of power increase is small, and thus a quasi-static approximation (in the power range) is valid. Because the new 8x8R fuel rod has a faster thermal time constant than the older types, and because the codes assume homogenized bundles, both the quasi-static procedure and the codes remain acceptable.

3.4.1.2 Transient Methods for Core Wide Events

Abnormal operational transients which effect the entire core are of safety concern only with regard to fuel rod overheating (CPR) and RCPB overpressurization considerations. Local (intra-assembly) peaking factors during core wide transients remain relatively low and essentially unchanged from normal operating values. Thus, local LHGR's do not closely approach the safety limit LHGR during such occurrences and are not a safety concern for initial or reload cores.

GETAB-SCAT Code Analysis

GE uses a framework of codes for predicting the hot bundle transient critical power ratio during core wide transient events. This framework has been consistently used by GE for initial and reload core licensing applications.

GETAB Transient Analysis

The central code in the GETAB transient analysis is the SCAT code[53], which incorporates the GEXL correlation[52] for predicting the change in bundle critical power ratio (CPR) during the transient. The SCAT code has been previously reviewed and approved by the staff in connection with transient CPR calculations of 7x7 and 8x8 bundles for ECCS Appendix K analyses[65]. The code is also considered to be acceptable for transient analysis applications. The two water rod 8x8R bundle geometry input for the SCAT code analysis does not represent a significant difference from fuel designs previously approved for analyses with the code (i.e., 7x7 or single water rod 8x8). The longer heated length (150 inches vs. 144 inches) and fuel rod diameter changes do not represent calculational difficulty, thus the 8x8R fuel element design is considered to be within the analysis capability of the code to yield conservative estimates of CPR.

The critical bundle power correlation used in the SCAT code analysis is the GEXL correlation. As discussed in Section 3.3.2.1, the GEXL correlation, employing the previously approved R-Factor formulation[53], results in non-conservative predictions of experimental CPR data for certain 8x8R local peaking factor distributions. However, these distributions are not expected to occur during the first operating cycle of the retrofit 8x8 assemblies. Thus, the use of the GEXL correlation for Cycle 3 of Peach Bottom 3 is acceptable. Additional data should be submitted to the staff for review, to justify the conservatism of the GEXL correlation for the second and subsequent cycles of operation of the retrofit 8x8 bundles.

Geometrical differences between the 8x8 and 8x8R fuel designs which can affect the bundle critical power calculation include the heated length, L, and thermal diameter, D_Q. The licensee was requested to provide additional information which would justify the acceptability of a single GETAB transient analysis for the two fuel designs for a given core wide transient event. The sensitivity results presented[5] show that there is a ΔCPR difference of approximately 0.001 between the two fuel geometries. Thus, we find it acceptable to perform a single GETAB transient analysis for both fuel types for a particular core wide event.

The effect of fuel densification on SCAT bundle critical power calculations has been considered. GE has presented analyses of the effect of densification power spikes on bundle critical power. These analyses utilized an "Integral Concept"[56]. The Integral Concept is widely used and considered to be an acceptable method for quantification of boiling transition correlations. The Integral Concept also requires an empirical base. This base has been found to conservatively represent BWR conditions by comparison with an independently established procedure[57]. GE has additionally demonstrated the effect of densification on R-factor and has concluded that the effect is insignificant. Based on the analyses presented, we find that the effects of fuel densification have been appropriately considered in the bundle CPR calculations.

GE develops the SCAT code initial conditions and transient history inputs from the nuclear analysis, core hydraulic analysis and plant system transient analysis. The Peach Bottom 3 inputs which do not vary from cycle to cycle appear in Table 5-3 of Reference 2. The remaining GETAB transient inputs were calculated for Reload 2 for each fuel type. The initial hot bundle flow for each fuel type is determined by the models and methods described in Section 4 of Reference 2. These methods are evaluated in Section 3.3.1 herein. The initial integral bundle power and local pin powers are determined by the GE BWR Simulator Code and Lattice Physics Methods, respectively. These codes and methods are evaluated in Section 3.2.1 herein.

Plant System Transient Analysis

GE develops the balance of the required input data for the GETAB transient (SCAT) code analysis from the output of the plant system transient (REDY)[58,59,60] code analysis. The plant system transient results required for each AOT event analyzed by the SCAT code consist of normalized core flow vs. time, reactor core pressure vs. time and core (hot bundle) nuclear power vs. time. These REDY code results are input into the GETAB analysis without modification (no conservatism factors applied to the output). Since safety analysis consequences (i.e., CPR, pressure increase) must be conservatively calculated, this would be an acceptable procedure provided the unmodified REDY code output is already adequately conservative, or provides for an overall adequately conservative CPR methodology. In this regard, the REDY code and related methods are currently under staff review and evaluation in connection with the conservatism afforded by transient predictions.

The REDY code by design is a best-estimate code. GE believes that adequate conservatism exists in the code predictions of plant system transient performance, by way of the conservatism factors applied to key nuclear (core) transient inputs. As seen in Table 5-2 of Reference 2, GE applies "design conservatism factors" (DCF's) of 0.95, 1.25 and 0.80 to the nominal Doppler, void and scram reactivities predicted by the nuclear analysis. These factors contribute to the currently used licensing basis analysis methods, and are intended to account for non-conservatisms and uncertainties associated with the calculation of the nuclear input parameters and the plant transient analysis models and methods.

Staff concern for the adequacy of the plant system transient methods has been raised by the apparently non-conservative predictions of transient tests recently conducted at a large BWR/4 reactor. The tests involved three end-of-cycle manual turbine trips, initiated from intermediate power levels with the direct (turbine stop valve position switch) reactor trip intentionally disabled. This required the reactor to trip on the indirect (high neutron flux) scram. Several key transient test parameters were underpredicted, even when the present licensing basis plant transient methods (REDY code and DCF's) were employed.

GE has evaluated[61] the differences between the turbine trip test conditions and the licensing basis event (turbine trip without bypass with a direct reactor scram) using a normalized REDY code model as well as a more detailed transient code model. The GE evaluation indicates that a degree of conservatism is available when using the licensing basis methods to predict the consequences of the licensing basis event. The staff agrees with this conclusion. The staff, in the interim, while reviewing another plant system transient code proposed by GE, has concluded that the present plant transient methods adequately predict the consequences of the limiting (licensing basis) core wide events[59].

Several of the plant system transient code models derive their input values from the fuel mechanical design. For example, the multi-noded thermal-hydraulic and heat transfer relationships utilize the fuel rod (fuel and clad) diameters and fuel column length as inputs. These parameters can, therefore, affect the dynamic behavior of the core via fuel thermal time constant and axial void sweep effects. When a substantial fraction of the core is composed of a mixture of fuel designs, the proper selection of the input values for fuel modeling must be carefully considered. The plant system transient code models heat transfer with a single fuel element representing the entire core. The staff has reviewed GE's analytical procedure for treating these fuel related inputs for the plant transient analysis of mixed cores such as Peach Bottom 3 Cycle 3.

GE's current procedure requires the single fuel element to be the "dominant" fuel type (except for fuel clad gap conductances) rather than a "weighted average." For Peach Bottom 3 this would result in the modeling of a 7x7 fuel element, since the 7x7 bundle is the dominant fuel type during Cycle 3. The 7x7 fuel element has a significantly slower fuel time constant compared with the 8x8 or 8x8R fuel element. Fuel time constant sensitivity studies with the REDY code(58) indicate that a faster fuel time constant results in more severe fuel consequences (e.g., peak heat flux). Thus, the present procedure may be somewhat non-conservative for mixed cores such as Peach Bottom 3. The staff is continuing to evaluate this GE analytical procedure for transient performance of mixed cores. Since the limiting transient event (which develops the operating limit MCPR's) for Peach Bottom 3 during Cycle 3 is not a core wide event (see Fuel Loading Error, Section 3.5.3), our concern pertaining to fuel element modeling for plant transient methods is not a significant concern for Cycle 3 of Peach Bottom 3. The staff is continuing its review of the current GE procedure on a generic basis, however.

GE also uses the REDY code predictions for evaluating conformance with the criteria relating to overpressurization of the reactor coolant system. REDY code simulations of the aforementioned transient tests (using the licensing basis DCF's) demonstrate that the peak transient pressure is consistently overpredicted by the code. The staff has considered the differences between the nature of the turbine trip tests and licensing basis pressurization events (i.e., turbine-generator trip without bypass with direct scram and Main Steam Isolation Valve Closure with indirect high flux scram) and concludes that the code can be expected to also overpredict the peak transient pressure due to the licensing basis event. The use of the REDY code is, therefore, considered acceptable for RCS overpressurization evaluations for Peach Bottom 3.

The Peach Bottom 3 REDY code input data, relating to pressure relief system characteristics, which do not vary from cycle to cycle appear in Table 5-1 of Reference 2. These characteristics are acceptable.

3.4.2 Transient Analysis Results

Reference 14 provides the results of the reanalysis of the most severe abnormal operational transients for Cycle 3. The types of abnormal operational transients analyzed were reactor pressure increase, feedwater temperature decrease, feedwater flow increase and local positive reactivity insertion events. The methods used in the analysis of the limiting transients applicable to Peach

Bottom 3 are described in Reference 2. Our evaluation of these methods is provided in Section 3.4.1. Our evaluation of the transient analysis results for Cycle 3, relative to the MCPR safety limit, LHGR safety limit and RCPB safety limit is provided in the following subsections.

3.4.2.1 Transients Affecting the Entire Core

Load Rejection Without Bypass

The load rejection without bypass transient produces the most severe reactor isolation. The reactor pressure increase due to fast closure of the turbine control valves causes a significant decrease in the core void fraction which in turn induces a positive core reactivity insertion, resulting in a rapid and substantial rise in the core neutron flux. The transient is terminated by a reactor trip initiated by fast closure position switches on the turbine control valves.

The analysis of this transient was performed at an exposure corresponding to EOC-3 and was done both with and without a recirculation drive motor trip. Since the severity of this event (reactor pressure increase and bundle CPR decrease) increases with burnup, the analysis provides conservative results for reactor operation for the entire cycle. The analyses were performed assuming an initial reactor thermal power level corresponding to 100% of the licensed limit, which is considered to be adequately conservative. The analysis results, provided in Section 9 of Reference 14, show that at the most limiting (EOC) condition, a 146 psi margin exists between the peak transient pressure and the 1375 psig RCPB safety limit.

The load rejection without bypass event also results in a significant reduction in MCPR from the operating value. This is caused by the combined effects of the rapid and substantial increase in the neutron flux, which results in a significant increase in the fuel rod surface heat flux together with the substantial increase in reactor pressure. The reduction in operating MCPR at EOC is 0.18 for the 7x7 fuel and 0.25 for the 8x8 and 8x8R fuel. Comparing these results with the other transient events affecting the entire core shows that the load rejection with bypass failure is the most severe core wide transient during Cycle 3.

Although the staff finds the calculation acceptable, the staff does not agree that the assumption at an initial reactor thermal power of 100% is adequately conservative. Because there is a 2% calorimetric power measurement uncertainty in plant operation, a reactor may possibly be operating at 102% power when the instrumentation reads 100%. Therefore, the staff normally requires that transient analyses be done assuming an initial reactor thermal power

>102%. Typically, BWR/4 plants have done this anlaysis at 104.5% (e.g., references 10 and 11), and early indication was given that the subject transient analysis would assume 3440 MWt which is 104.5% of rated thermal power (reference 18, Table 5-6, p. 5-64). Moreover, no justification for the change has been submitted. Therefore, the staff finds this calculation to be adequately conservative only for power levels of 98% and below.

This non-conservatism does not affect the conclusions fo the transient analysis evaluation because of the following:

(1) The overpressurization analysis of MSIV closure with indirect scram (discussed in Section 3.6) shows adequate margin to the 1375 psig pressure limit. Experience has shown (reference 2, p. 5-25) that this event is slightly more severe than turbine trip without bypass. Therefore, there is adequate assurance that the pressure limit will not be violated by a load rejection without bypass.

(2) The reduction in critical power ratio is significantly greater for a fuel loading error than for any of the core-wide transients, as discussed in Sections 3.5.3 and 3.4.3. Therefore, the load rejection without bypass is not limiting provided the plant is restricted by the operating MCPR limits given in Table 3.3.

Other Core Wide Transients

The other core wide transients analyzed for Cycle 3 were feedwater controller failure (maximum demand) and loss of 100°F feedwater heating (LFWH) capability. The event descriptions for these transients are given in Reference 2. These analyses assumed an initial reactor thermal power of 104.5% of rated.

A comparison of these events with the load rejection without bypass shows that the reactor pressure increase associated with these two transients is less severe than the pressure increase for the load rejection without bypass. The ΔCPR's for the load rejection without bypass event are more severe for all fuel types at all exposures.

3.4.2.2 Rod Withdrawal Error

The rod withdrawal error (RWE) transient can occur when the reactor operator makes a procedural error and attempts to withdraw the maximum worth control rod to its fully withdrawn position. The attendant local power increase in the fuel assemblies in the vicinity of the withdrawn control rod causes a reduction in the bundle CPR's in addition to an increase in the fuel rod local LHGR's. The information provided in Reference 2 indicates that the local power range monitor subsystem (LPRM's) will detect and alarm a high local power condition. However, even if the reactor operator ignores the LPRM alarm, References 1 and 2 indicate that the rod block monitor subsystem (set at 107% of rated power at 100% core flow) will terminate the RWE transient with the control rod only 4.5 feet withdrawn. This will limit the critical power ratio to 1.27 for the effected 7x7 assemblies and 1.21 for the 8x8 and 8x8R assemblies.

A RBM rod block occurring at 107% power and full core flow results in peak linear heat generation rates of 16.0 and 14.5 Kw/ft for the affected 7x7 and 8x8/8x8R assemblies, respectively. These calculated LHGR's are below the safety limit LHGR's for 7x7 and 8x8 fuels even when the effects of densification spiking are included and are therefore acceptable to the staff.

3.4.3 MCPR Operating Limits for Rated Conditions

Abnormal operating transients, as discussed in the previous section, will reduce fuel bundle critical power ratios from steady-state operating values. In order to assure that the 1.07 fuel cladding integrity safety limit MCPR is not violated during the most severe transient, the most limiting transients have been reanalyzed for Cycle 3 to determine which transient event results in the largest decrease in critical power ratio (i.e., ΔCPR). The most limiting abnormal operational transient which can occur at any time during Cycle 3 is the load rejection without bypass. A summary of the calculational fuel type dependent ΔCPR's, for the exposure increments analyzed,[14] is as follows:

TABLE 3.1

Fuel Type	ΔCPR
7x7	0.18
8x8	0.29
8x8R	0.25
PTA (one 8x8 test assembly)	0.26

Addition of the above CPR's to the safety limit MCPR, would normally provide the minimum operating limit MCPR, for each fuel type (and exposure increment) required to avoid violation of the safety limit, should the most limiting transient occur. The licensee has therefore proposed the following MCPR operating limits:

TABLE 3.2

Fuel Type	MCPR
7x7	1.25
8x8	1.32
8x8R	1.32
PTA	1.33

However, the licensee reports in the amended reload supplement(14) that the worst case fuel loading error (FLE), consisting of a fresh 8x8R bundle misoriented in its correct 8x8R cell location, results in a MCPR of 1.03 when starting from an initial MCPR of 1.32. Furthermore, the licensee reports(6,7) that placing a fresh 8x8R in an exposed 7x7 location results in a 1.04 MCPR. Finally, no violation of the safety limit MCPR occurs when a fresh 8x8R assembly is placed into a standard 8x8 cell location.

As discussed in Section 3.5.3, the staff has the fuel loading error under generic review. Until the issues raised in connection with this event are resolved, the licensee has agreed to increase sufficiently to account for the possibility of a fuel loading error such that the safety limit MCPR is not violated. Thus, based on the analysis of both the most severe abnormal operational transients and the fuel loading error, we have determined that the operating limit MCPR's for Peach Bottom 3 during Cycle 3 be as follows:

TABLE 3.3

Fuel Type	MCPR
7x7	1.28
8x8	1.36
8x8R	1.36
PTA	1.33

3.4.4 MCPR Operating Limits for Less than Rated Flow

To assure that the 1.07 safety limit MCPR is not violated for the limiting flow increase transient (recirculation pump speed control failure) starting from less than rated flow conditions, the licensee will operate Peach Bottom 3 in conformance with the limiting conditions for operation as stated in paragraph 4.5-K of the Technical Specifications. This requires that for core flow rates less than rated flow, the licensee shall maintain the MCPR above the minimum operating values. The minimum MCPR values for less than rated flow are equal to the MCPR for rated flow multiplied by the respective K_f reactor values appearing in Figure 3.5.1-E of the Technical Specifications. The K_f factor curves were generically derived and assure that for the most limiting (flow increase) transient, occurring from less than rated core flow, the actual MCPR will not exceed the safety limit MCPR of 1.07. The K_f curves were generically derived(58) and are applicable for all fuel types present in the Peach Bottom 3 Cycle 3 core.

Application of the above stated K_f factors, for reduced flow conditions, results in calculated consequences for the limiting anticipated flow increase transients which do not exceed the thermal limits of the fuel.

Thus, we conclude the analyses and the operating limits, based upon the use of the General Electric Thermal Analysis Basis(56), have been conservatively applied to Peach Bottom 3, Reload 2, and are acceptable.

3.5 Accident Analysis Evaluation

3.5.1 Loss of Coolant Accident

ECCS Appendix K Model Applicability

Because of the physical differences between the standard 8x8 (and 7x7) and the retrofit 8x8 fuel designs described in Section 3.1.1, we reviewed the acceptability of continued application of the previously approved,(65) unchanged, ECCS-LOCA models to the new fuel. Our review and evaluation of GE's responses(62) to our request for justification of such continued application follow.

The staff agrees with the following assertions made by GE:

- All parameters of the new 8x8R fuel, such as hydraulic diameter, pressure, flow, power, and temperature are within the range of data used in developing the GEXL correlation in the ECCS-LOCA models to determine time-to-DNB for the retrofit fuel is acceptable. Also, the R-Factors used in this (LOCA) application of GEXL result from a conservative and therefore acceptable initialization procedure.

- Slightly higher PCT's are calculated for the new fuel (compared to the standard 8x8 fuel at the same MAPLHGR). This is due to the small change in fuel dimensions (resulting in reduced surface area) and a shift in local power peaking toward the center of the bundle. These effects are properly included in the models, so continued application of the models in the new PCT-MAPLHGR range is acceptable.

- GE has previously stated that substantial changes in rod dimensions, spacing, linear heat generation rate, and lattice design do not significantly affect spray cooling heat transfer coefficients. We agree with GE that the changes from the standard 8x8 fuel design to the two water rod retrofit fuel design will not affect the overall conservatism and acceptability of the spray cooling coefficients assumed for the new fuel in the ECCS model inputs.

- The radiative heat transfer model used in the CHASTE code was written to handle calculations with various size rods, including rods of unequal radii. Hence it is capable of calculating radiative heat transfer for the new fuel design, and its application for that purpose is acceptable.

- The data base used to develop the swelling and rupture model covered the range of internal pressures and temperatures expected for the new retrofit 8x8 fuel. The swelling and rupture model is therefore equally acceptable to both the standard and retrofit 8x8 fuel designs.

- The data base used to develop the gap conductivity model included the range of temperature, internal pressure, and gap size applicability to the retrofit fuel design. Application of the gap conductivity model to the new fuel is therefore acceptable.

- It has been known by the staff that GE's method of initializing gap conductivity (as a function of assumed fuel rod linear power level) is not the most conservative possible initialization method. However, GE has shown that this initialization method, when applied to the retrofit 8x8 fuel, is slightly more conservative than when it is applied to the standard 8x8 fuel design, where its application has previously been accepted. The initialization method therefore is also acceptable for use with the new retrofit fuel.

- The retrofit 8x8 fuel has a more uniform axial power profile and a six-inch longer active fuel length. These factors make it possible that the plane of maximum PCT could shift to a higher elevation (power is lower above the core midplane, but loss-of-nucleate boiling occurs earlier for the new fuel compared with the old fuel). However, the application of the model to Peach Bottom 3 includes a calculation to demonstrate that such a shift has not occurred[63]. Continued use of that calculation provision for Peach Bottom 3 will ensure that the application of the model to the new fuel will be at the axial plant producing the highest PCT, and will therefore be acceptable.

For the reasons stated above, we conclude the continued application of the present GE ECCS-LOCA ("Appendix K") models to the 8x8 retrofit fuel is acceptable for Peach Bottom 3.

<u>Small Break Analysis</u>

The licensee has proposed[1] to raise the setpoints of the 11 safety/relief valves by approximately 2.3% to provide increased simmer margin. This change has no effect on the large-break LOCA analyses because of the rapid depressurization involved. However, for a small break, system pressure initially rises after the MSIVs close. Therefore, the effect on high-pressure coolant injection and on automatic depressurization was examined.

One of the requirements for the HPCI (and RCIC) systems is that they be capable of providing the design flow at the lowest safety/relief valve setpoint (now 1105 psig + 1% uncertainty = 1116 psig). The licensee has found[14] that the Peach Bottom 2 HPCI and RCIC systems will still meet the design requirements at this slightly higher pressure. The staff finds this acceptable.

The small-break analysis assuming single-failure of the HPCI system and initiation of the ADS is slightly sensitive to the safety/relief valve setpoints. The staff has previously examined this effect as part of another review, and has concluded that the small change in peak cladding temperature (I∿40°F) will not make the small break limiting.

ECCS Appendix K LOCA Analysis Results

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 core reloading "...the licensee shall submit a reevaluation of ECCS performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR 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 results and assumptions.

In December of 1976, the NRC staff was informed that certain input errors and computer code errors had been made in the evaluations that were provided under the requirements described above. An Order was issued by the Nuclear Regulatory Commission to the Philadelphia Electric Company on March 11, 1977, requiring that corrected "Revised calculations fully conforming to the requirements of 10 CFR 50.46 are to be provided for Peach Bottom Atomic Power Station Unit 3 as soon as possible." Such corrected analyses were provided for the previous core and the reloaded (Cycle 3) core in Reference 8. The revised calculations included corrections of all of the input errors and all computer code errors. The corrected analyses were performed using a calculational model which contains several model changes approved by the staff in its Safety Evaluation issued April 12, 1977.

We have reviewed the corrected analyses submitted in Reference 8, and the resulting Technical Specification changes submitted in Reference 1. We conclude that the Peach Bottom Atomic Power Station Unit #3 (PB#3) will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: 1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Figures 3.5.1.A,B,C,D,E,F, and G of Reference 1; and 2) when it is operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss-of-Coolant-Accident.

The analyses submitted in Reference 8 provide all information requested in the NRC letter to GE on June 30, 1977 regarding number of breaks to be analyzed, documentation to be provided, etc, for the new analyses. These analyses for PB#3 reference the lead plant (James A. FitzPatrick Nuclear Power Plant) analyses for BWR/4 plants with the low-pressure-coolant-injection system modification[65]. The following description is provided of particular features of the analyses which are different from the lead plant, and the reason underlying those differences.

The break spectrum (i.e., PCT vs. break size) for the lead plant showed that the particular break producing the highest PCT for the lead plant was a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break.[65] However, the break spectrum for PB#3 showed that the particular break producing the highest PCT is the largest (100%) discharge line break.

The SER for the lead plant[66] explains the reasons why the discharge break location is limiting for that plant. As explained more fully in that SER, the largest break in the largest pipe would normally be expected to be limiting (the largest pipe is the suction pipe). However, the LPCI modification (also explained more fully in the lead plant SER) results in at least one loop of the LPCI system being available to help mitigate the consequences of suction pipe breaks even with the worst assumed single failure; but, due to certain piping and valve locations, with certain single failure assumptions, no LPCI system is available for the smaller, discharge line break. This results in a "tradeoff" or "compensating effects" situation where a larger, normally more severe break (suction line) has more ECCS available to mitigate its consequences, while a smaller, normally less severe break (discharge line) has less ECCS. The lead plant SER states that in most cases this "tradeoff" results in the discharge location being limiting, as it is for FitzPatrick and PB#3.

In order to justify that the largest discharge line break is limiting for PB#3, it is necessary to determine that no discharge or suction break size that was not specifically analyzed for PB#3 could be more limiting than the discrete sizes that were specifically analyzed.

The same arguments presented in the lead plant SER[86] regarding PCT vs. break size also apply to PB#3. For PB#3 the uncovery-time-interval vs. discharge break area curve peaks at 66% of the largest discharge line break's area. For suction breaks, the uncovery-time-interval vs. suction break area curve peaks at 100% of the largest suction line break's area.

Uncovery-time-interval is generally the single most important "time" in determining ultimate PCT. However, two other times that significantly affect PCT are departure-from-nucleate-boiling (DNB) time and uncovery time. Both of these times occur earlier as break size is increased; earlier DNB and earlier uncovery times each cause PCT to increase due to earlier loss of heat removal capability.

Therefore, for suction line breaks, all three "times" (uncovery-time-interval, DNB time, and uncovery time) are each individually at their value which would cause highest PCT at the same size (largest) break. Thus the largest suction line break would clearly have the highest

PCT of any suction line break. This largest suction break's PCT was calculated to be 2187°F for PB#3. For discharge line breaks, one of these "times" (uncovery-time-interval) would tend to cause the highest PCT at 66% of the largest discharge line's break area; the other two times (DNB and uncovery) would tend to cause highest PCT for 100% of the largest discharge line's break area. Specific calculations for these two breaks for PB#3 have shown the "66%" break's PCT to be 2188°F, and the "100%" discharge line break's PCT to be 2198°F.

As illustrated in Figure 6a of Reference 8, the uncovery-time-period vs. discharge break area curve peaks very sharply at "66%"; and change to a slightly larger or smaller break area would cause a shift to a significantly shorter uncovery-time-period which would overcompensate for any possible effects on PCT in the other direction due to the size change (i.e., changes in DNB time or uncovery time). Between 80% and 100% the uncovery-time-period increases and the break at 100% results in the largest period for which the hot node remains uncovered. Over this range the 100% break results in the highest calculated PCT since, if two breaks have similar times for which the hot node remains uncovered, then the larger of the two breaks will be limiting since it would have an earlier uncovery and earlier DNB time (i.e., the larger break would have the more severe blowdown heat transfer analysis).

We therefore conclude, for the reasons stated above, that the most limiting break is the largest discharge line break for PB#3. That break was used to generate the above referenced MAPLHGR limits, which we therefore find acceptable as stated previously.

3.5.2 Steamline Break Accident

The radiological consequences of a postulated steamline break outside of the primary containment are dependent on the amount of primary coolant lost during the accident and the concentration of the radioactivity in the coolant. The amount of coolant lost is primarily a function of plant system parameters, which would be insignificantly changed by introduction of the 8x8R fuel assemblies into the core. The concentration of radioactivity in the coolant is limited by the plant Technical Specifications and is also unchanged for this reload. Therefore, the previously calculated radiological consequences of a postulated steamline break accident are unaffected by the use of the 8x8R fuel assemblies.

3.5.3 Fuel Loading Error

Reference 14 gives the results of the fuel loading error analysis for Cycle 3. The most severe fuel loading error event consists of a misoriented, fresh 8x8R fuel bundle. The information in Reference

14 indicates that this worst case fuel loading error, were it to occur, would result in a minimum critical power ratio (MCPR) of 1.03 in the misoriented fuel bundle during steady-state full power operating conditions. Fuel bundles adjacent to the misloaded fuel-assembly would be negligibly affected by the misoriented bundle. The calculated MCPR of 1.03 in the misloaded bundle violates the 1.07 fuel cladding integrity safety limit MCPR.

The fuel loading error event is being generically reviewed by the staff and a generic resolution is anticipated. Our ongoing review includes an evaluation of the adequacy of proposed new fuel loading error methods, event probabilities resulting from improved core loading control procedures, in addition to acceptable consequences for the fuel loading error event. Until these evaluations are complete the licensee committed to increase the MCPR operating limits to values which will assure that, during normal operation, the safety limit MCPR will not be violated. For Cycle 3, the MCPR operating limits shown in Table 3.3 herein will assure that the most severe fuel loading error will not cause a violation of the safety limit MCPR.

3.5.4 Control Rod Drop Accident

The postulated control rod drop accident assumes that a control rod has been fully inserted and becomes stuck in this position. The control rod drive is assumed to be uncoupled and withdrawn. The rod subsequently becomes free and rapidly falls out of the core onto the withdrawn drive coupling. The amount of reactivity represented by this event is introduced into the reactor core at a rate consistent with the maximum control rod drop velocity.

There are two criteria which must be satisfied in the analysis of the control rod drop accident:

- Reactivity excursions must not result in a fuel enthalpy greater than 280 cal/g at any axial pellet location in any fuel rod. This limit assures that dispersal of fuel into the reactor coolant will not occur.
- The maximum reactor pressure during any portion of the accident must be less than the value that will cause reactor system stresses to exceed the emergency condition stress limits defined in the ASME code.

It has previously been demonstrated[67] that unless there is dispersal of hot fuel into the coolant, the pressure surge is not significant. Therefore, the evaluation of the rod drop accident is concerned primarily with the 280 calorie/scram limit.

The analysis of the control rod drop accident was performed by General Electric on a generic (bounding) basis and presented in Reference 2. The bundle cross sections, developed by the lattice calculations (discussed in Section 3.2.1) for the rod drop excursion model, are homogenized. As a result, the rod drop excursion model does not recognize the difference between 7x7, 8x8 or 8x8R fuel. Therefore, the calculational model used in the generic analysis remains acceptable for the new fuel design. The evaluation of the control rod drop accident thus consists of ensuring that the appropriate parameters of the new core are bounded by the input parameter values used in the generic analysis.

The generic analysis assumes the slowest scram allowed by the Technical Specifications (and assumes that the dropped rod does not scram), the most rapid credible rod drop velocity, and the smallest (i.e., high exposure) value for delayed neutron fraction. The remaining parameters of interest include the Doppler feedback, the scram reactivity, and the accident reactivity characteristics.

We have reviewed the bounding calculations presented in Reference 1 with regard to the 280 cal/g limit and find them to be acceptable for reference, provided the key input parameters for the Peach Bottom 3 Reload 2 application fall conservatively within the assumed bounding analysis values. The key parameters are Doppler coefficient, scram reactivity function and accident reactivity function.

The licensee has compared the Doppler coefficient as a function of fuel temperature used in the generic analysis with that calculated for the upcoming cycle core. This comparison was done for both cold and hot moderator conditions. It was found that the Doppler coefficient for the upcoming cycle core is more negative than the corresponding coefficient used in the generic analysis for all fuel and moderator temperatures. Therefore, the Doppler coefficient is properly bounded.

The licensee has compared the scram reactivity curve (negative reactivity vs. time) with that assumed in the generic analysis for both cold and hot moderator conditions. It was found that the scram curve for the upcoming cycle inserts more negative reactivity than that assumed in the generic analysis. Therefore, the scram reactivity function is properly bounded.

The licensee has calculated the worst-case accident reactivity characteristics (i.e., positive reactivity inserted vs. dropped-rod position) consistent with the constraints imposed by the Rod Sequence Control System (RSCS) for both cold and hot startup conditions. The curve in the generic analysis to which this must be compared depends on the maximum local peaking factor in the four bundles surrounding the dropped rod. Attention must be paid to the local peaking factor in this fashion because the calculational methods use homogenized bundles, while the limiting parameter is applied to one fuel rod. The licensee used the generic curve corresponding to a local peaking of 1.3. This is acceptable because 1.3 bounds the local peaking factors for all three fuel types for the entire cycle.

It was found that the accident reactivity characteristic calculated for the upcoming cycle for cold startup conditions bounded by (i.e., is less than) the curve used in the generic analysis except for approximately the first two feet, where the curve for the upcoming cycle is slightly ($\sim$0.0005 Δk) greater than the generic analysis curve. However, the curves cross at $\sim$0.35% Δk, well before prompt criticality occurs. Since the analysis of the rod drop accident is sensitive primarily to the prompt excursion and relatively insensitive to the long-period effects of the early portion of the accident reactivity curve, the staff agrees that the slight excess at the beginning of the curve will have an insignificant effect on the analysis. The curve for hot startup conditions is entirely bounded. Therefore, the entire accident reactivity characteristic is effectively bounded, and the generic analysis is applicable. Since the generic analysis shows the 280 calorie/gram criterion to be met, it is concluded that the analysis of the rod drop accident for the upcoming cycle is acceptable.

3.5.5 Fuel Handling Accident

The refueling accident has been generically reanalyzed[2] to determine the radiological consequences for the 8x8R fuel assembly. The analysis assumes (1) the fuel assembly is dropped from the maximum height (maximum potential energy) allowed by the fuel handling equipment, (2) none of the kinetic energy is viscously dissipated as the assembly falls through the water covering the core and (3) none of the kinetic energy is absorbed by the fuel material (UO_2) in the assembly. Using energy methods to predict cladding failures, it is shown that a total of 125 8x8R fuel rods fail during the accident. This compares with 111 rods for a 7x7 core. There would be no difference in failed rods for an 8x8 core. The evaluation also conservatively assumes that the fractional plenum activity in the 8x8R rod is the same as for a 7x7 rod. In actuality an 8x8R rod would have substantially lower gap activity as compared to a 7x7 rod as a result of the significantly lower linear heat generation rate (fuel temperatures) applicable to the new fuel bundle design. Comparing the average activity per 8x8R fuel rod to the average activity per 7x7 rod together with the number of failed rods for each bundle type (125 vs 111), it is shown by the licensee that the 8x8R fuel bundle results in a relative activity release of only 88% of the activity released for a 7x7 core. Thus, of the total activity available for release above the core, the fission product activity component attributable to the fuel is less for the 8x8R fuel than for the 7x7 fuel. The FSAR analyses of the 7x7 core showed fuel handling accident dose consequences which were well within the guidelines set forth in 10 CFR 100. Thus, we conclude that the dose consequences of the fuel handling accident associated with the 8x8R fuel assembly for Peach Bottom 3, are also well within 10 CFR 100 guidelines and are acceptable.

3.5.6 Recirculation Pump Seizure Accident

The analysis of the single pump seizure event shows that it is relatively mild with regard to radiological consequences, plant system behavior and fuel performance when compared to a large LOCA. For both accidents recirculation flow rapidly terminates. In the case of the LOCA, the forced recirculation flow disruption is more rapid and severe than the pump seizure event. Furthermore, the loss of coolant accident results in core uncovery with subsequent rapid and substantial temperature rise of the fuel cladding. The pump seizure accident also does not result in as rapid and core pressure drop as does the LOCA. The combination of higher peak cladding temperature and lower RCS pressure during a LOCA event results in greater cladding perforation potential for the LOCA than the pump seizure event. The staff agrees that the potential adverse effects on the fuel of a pump seizure accident are conservatively bounded by a LOCA. Additionally, the LOCA results in the removal of the reactor coolant pressure boundary as a barrier to the release of fission products outside of containment. The single pump seizure does not result in the loss of this barrier. Therefore, it may also be concluded that the radiological consequences associated with the LOCA conservatively bound the radiological consequences of the pump seizure event. Since the radiological consequences of the LOCA as described in the Peach Bottom 3 FSAR were shown to be acceptable, the consequences of the pump seizure accident are also considered to be acceptable.

3.6 Overpressurization Analysis

For Cycle 3, the licensee proposed to raise the safety/relief valve setpoints approximately 2.3% to increase the simmer margin. The[14] licensee presented the results of an overpressurization analysis to demonstrate that margin exists to the ASME code allowable reactor vessel pressure limit. This limit, as discussed previously in

Section 3.4, is 100% of the vessel design pressure and corresponds to a pressure of 1375 psig. The methods(2) used for this analysis are evaluated in Section 3.4.1.2 herein. The transient event analyzed was the rapid closure of all main steam isolation valves (MSIV) with an indirect reactor trip on high neutron flux. The analysis was performed assuming an initial core thermal power level corresponding to 104.5% of the license limit. In addition, the analysis conservatively utilized the end-of-cycle scram reactivity insertion rate curve, with void and Doppler reactivity coefficients applicable for this reload. Moreover, no credit was taken for the relief function of the 11 dual action safety/relief valves installed on the main steam lines. All valves were assumed operative in the analysis. The result of the analysis shows that the peak pressure at the bottom of the reactor vessel is 1301 psig.

Furthermore, generic analyses(2) applied to Peach Bottom 3 show that the failure of one of the safety/relief valves would cause the maximum vessel pressure to increase by no more than 20 psi. Thus, the peak transient pressure at the vessel bottom for the MSIV closure overpressurization event from full power with flux scram, no relief function of the safety/relief valves and one failed safety/relief valve is calculated to be less than 1321 psig. This results in an adequate margin to the 1375 psig ASME code allowable pressure limit and is thus acceptable to the staff.

4.0 Physics Startup Testing

As part of our evaluation of Reload 2 of Peach Bottom 3, we reviewed the physics startup test program which will be conducted by the licensee at the beginning of Cycle 3. The test program description is provided in Reference 2. Based on our review of the information(2,5) provided by the licensee, the staff finds that the physics startup tests together with the tests required to assure compliance with the Technical Specifications, provide an acceptable physics startup test program.

5.0 Technical Specification Changes

The proposed revisions(1) to the Technical Specifications for Cycle 3 operation include changes to the MCPR safety limit, the MCPR operating limits, the MAPLHGR vs. planar average exposure curves and the safety/relief valve setpoints. The bases for the Technical Specification changes are documented in the reload submittals provided by the licensee. The bases for the proposed revisions have been evaluated by the staff and are discussed in Section 3.0.

As discussed in Section 3.3.2.1 of this evaluation, the MCPR safety limit has been increased from 1.06 to 1.07 for Cycle 3. This is to accommodate the combined effects of the flatter intra-assembly power peaking distribution associated with the retro-fit 8x8 reload fuel assembly and the revised core relative bundle power histogram (distribution) associated with a reload 8x8R cycle approaching equilibrium conditions. Based on our review of the information submitted by the licensee, the staff finds the 1.07 safety limit MCPR proposed for Peach Bottom 3 during Cycle 3 to be acceptable.

The Peach Bottom 3 Technical Specification revisions for Cycle 3 also address a change in the MCPR operating limits for the 7x7, 8x8 and 8x8R fuel types, based on the reload safety analysis results presented in Reference 14. As discussed in Sections 3.4.3 and 3.5.3 herein, the proposed operating limit MCPR's must be increased in some cases to assure that the 1.07 safety limit MCPR is not violated in the event of a fuel loading error. The Technical Specification operating limit MCPR's for each fuel type and exposure interval selected by the licensee must, therefore, be those appearing in Table 3.3 of this evaluation. The adjusted MCPR operating limits have been discussed with and accepted by the licensee.

The licensee has also proposed changes and additions[1.6] to the MAPLHGR vs. planar average exposure curves currently appearing in the Technical Specifications. Our evaluation of the proposed curves is discussed in Section 3.5.1. The MAPLHGR changes for the 7x7 and standard 8x8 fuel types reflect the results of revised LOCA calculations, performed to correct all of the GE input errors, made in connection with the previous LOCA analysis. An additional MAPLHGR curve, based on corrected inputs for the reload 8x8R fuel is also provided. Based on our evaluation of the information provided, the staff finds the proposed new MAPLHGR vs. planar average exposure curves appearing in Figure 3.5.1 of Reference 1 to be acceptable.

6.0 Environmental Consideration

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.

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: May 17, 1978

8.0 References

1. "Application for Amendment of Facility Operating License DPR-56," submitted as an Enclosure of Philadelphia Electric Company letter (Eugene J. Bradley) to NRC (Director of Nuclear Reactor Regulation) dated December 19, 1977.

2. "General Electric Boiling Water Reactor Reload-2 Licensing Application for Peach Bottom Atomic Power Station Unit 3, "NEDO-24039, submitted as an Enclosure of Philadelphia Electric Company letter (John Hankins) to NRC (George Lear) dated August 30, 1977.

3. Philadelphia Electric Company letter (Eugene J. Bradley) to NRC (Director of Nuclear Reactor Regulation) dated January 17, 1978.

4. Philadelphia Electric Company letter (M. J. Cooney) to NRC (George Lear) dated February 2, 1978.

5. "Responses to Request for Additional Information," submitted as an Enclosure of Philadelphia Electric Company letter (Eugene J. Bradley) to NRC (George Lear) dated February 17, 1978.

6. Philadelphia Electric Company letter (Dalhoff) to NRC (Director of Nuclear Reactor Regulation) dated May 8, 1978.

7. Philadelphia Electric Company letter (Dalhoff) to NRC (Director of Nuclear Reactor Regulation) dated May 11, 1978.

8. "Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 3," NEDO-24082, submitted as an Enclosure of Philadelphia Electric Company letter (Eugene J. Bradley) to NRC (Director of Nuclear Reactor Regulation) dated December 19, 1977.

9. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 52 to Facility Operating License No. DPR-57, April 11, 1978.

10. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 46 to Facility Operating License DPR-46, May 2, 1978.

11. Safety Evaluation supporting Reload-1 License Amendment No. 33 for Peach Bottom Atomic Power Station Unit No. 3, dated March 23, 1977.

12. BWR/6 Fuel Design, General Electric Report, NEDE-20948P, dated December 1975.

13. "Report to the Advisory Committee on Reactor Safeguards in the Matter of Georgia Power Company et. al. on Edwin I. Hatch Unit 2," dated January 4, 1978.

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18. "Generic Reload Fuel Application", General Electric Report, NEDE-24011-P, May 1977.

19. "Creep Collapse Analysis of BWR Fuel Using SAFE-COLAPS Model", NEDE-20606PA, August 1976.

20. NRC letter (W. Butler) to General Electric (I. Stuart), April 2, 1975.

21. General Electric Report, "Generic Reload Fuel Application, "NEDE-24011P-1, page B-21.

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23. 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, dated April 1975.

24. "GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods", General Electric Report, NEDC-20181, November 1973.

25. "Technical Report on the General Electric Company 8x8 Fuel Assembly" by the Regulatory Staff, February 5, 1974.

26. Philadelphia Electric Company letter (M. Cooney) to NRC (G. Lear) dated July 5, 1977.

27. General Electric letter (G. Sherwood) to NRC (D. Eisenhut), "NRC Questions on Rod Bowing", March 29, 1977.

28. "BWR/6 Fuel Design, Amendment No. 1", NEDE-20948-1P, November 1976.

29. "BWR/4 and BWR/5 Fuel Design, Amendment No. 1", NEDE-20944-1, January 1977.

30. USAEC letter (V. Moore) to General Electric (I.S. Mitchell), "Modified GE Model for Fuel Densification", Docket 50-321, March 22, 1974.

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56. O. Glenn Smith, W. M. Rohrem, Jr., and L. S. Tong, "Burnout in Steam Water Flow with Axially Nonuniform Heat Flux", ASME Paper 65-WA/HT-33, November 1965.

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64. Staff Safety Evaluation supporting Amendment 36 to Operating License DPR-44, Peach Bottom Atomic Power Station Unit 2, dated August 16, 1977.

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67. GE report, "The Mechanical Effects of Reactivity Transients," APED-5455, January, 1968.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 41 to Facility Operating License No. DPR-56 issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications for operation of the Peach Bottom Atomic Power Station Unit No. 3. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications for the Peach Bottom Atomic Power Station, Unit No. 3 to: (1) permit operation of the facility during Cycle 3 with up to 252 improved two water rod 8x8R reload fuel bundles, designed and fabricated by the General Electric Company and having an average enrichment of 2.23 wt/% ^{235}U, and (2) revise the Maximum Average Planar Linear Heat Generation Rates as determined by the reevaluation of the ECCS performance.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on February 2, 1978 (43 FR 4468). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated December 19, 1977, as supplemented August 30, 1977, January 17, February 2 and 17, May 8 and 11, 1978, (2) Amendment No. 41 to License No. DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 17th day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors