UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

STAFF EVALUATION OF THE INDIVIDUAL PLANT EXAMINATION

LICENSE NOS. DPR-44 and DPR-56

PECO ENERGY COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-277 AND 50-278

## I. INTRODUCTION

In August of 1992, the Philadelphia Electric Company submitted the Peach Bottom Atomic Power Station Units 2 & 3 Individual Plant Examination (IPE) in response to Generic Letter (GL) 88-20 and associated supplements. Because Peach Bottom Unit 2 has already been analyzed by the NRC in the NUREG-1150 study, the staff modified its "Step 1" IPE submittal review procedure; for Peach Bottom, the contractor's review is based on a comparison between the results of the IPE submittal and the results of the NUREG-1150 study (which is documented in NUREG/CR-4550 and NUREG/CR-4551, Level 1 and 2 analyses, respectively). Unlike the other plants, it was not necessary to send requests for additional information (RAIs) to the Peach Bottom licensee during the staff's review.

The modified Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore the contractor's, Sandia National Laboratories (SNL), review considered (1) the completeness of the information and (2) the reasonableness of the results given the Peach Bottom Units 2 & 3 design and operation. The results of SNL's "modified Step 1" review and the staff's conclusions are discussed in this Staff Evaluation (SE). SNL's Technical Evaluation Report (TER) is attached.

## II. EVALUATION

The Peach Bottom Atomic Power Station Units 2 & 3 are General Electric BWR-4 reactors with a Mark I containment. It is stated in the submittal that "Unit 3 is essentially identical (except for differences due to timing of design changes) to Unit 2, and therefore the conclusions of this report are valid for both units based on a review of the differences in PRA modeled systems." It is also stated that "The Level 1 and Level 2 Peach Bottom PRA analyses were performed based on the current Unit 2 design and confirmed to be representative of Unit 3." Therefore, the Peach Bottom IPE analysis is based on Unit 2 plant design. When evaluating the Unit 2 response to initiating

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events, the licensee assumed that both units were operating at full power. Shared components between the units were explicitly modeled; loss of offsite power was modeled as placing simultaneous demands on the shared systems; all other initiating events were modeled with the assumption that Unit 3 is at full power.

The IPE has estimated a core damage frequency (CDF) of 6E-6/reactor-year from internally initiated events; this value includes a contribution of 1E-7/reactor-year from internal floods. Categorized by general accident class, loss of offsite power (LOSP) contributed 25 percent of the total CDF, station blackout 9 percent, anticipated transients without scram (ATWS) 26 percent, transients 27 percent, and loss of coolant accidents (LOCAs) 11 percent. Other general accident classes each contributed less than 5 percent. The important system/function contributions to the CDF, in a decreasing order of importance, are high pressure coolant injection (HPCI) turbine fails to start, reactor core isolation cooling (RCIC) turbine fails to start, HPCI turbine fails to run, and RCIC turbine fails to run.

The mean CDF of 6E-6/yr is comparable to the mean value of 5E-6/yr for the NUREG/CR-4550 (NUREG-1150 plant) analysis of Peach Bottom, Unit 2. Station blackout (9 percent or 5E-7/yr) is not a dominant contributor to CDF in the IPE submittal compared with the NUREG/CR-4550 Peach Bottom, Unit 2, analysis (50 percent or 2E-6/yr). On the other hand, LOSP excluding blackouts (25 percent or 1E-6/yr), is a dominant contributor to CDF in the IPE submittal in contrast to the NUREG/CR-4550 analysis (2 percent or 9E-8/yr). The differences between the two studies are not unexpected and can be explained in terms of modeling and data differences. For example, it appears that sequences that were labeled as station blackout in one analysis were labeled as LOSP transients in the other analysis because of modeling differences (e.g., credit taken in the IPE for cross-tie EDGs but not in NUREG/CR-4550). Since the total frequency of core damage from LOSP plus blackout sequences is about the same for both studies, the result appears reasonable. Also, the IPE used more recent and more plant-specific data than the NUREG/CR-4550 study.

Transients are a more dominant contributor to CDF (27 percent or 2E-6/yr) in the IPE submittal than in the NUREG/CR-4550 analysis of Peach Bottom, Unit 2, (3 percent or 2E-7/yr). LOCAs are more than twice as important in the IPE (11 percent or 6E-7/yr) as in NUREG/CR-4550 (6 percent or 3E-7/yr) apparently due to the higher initiating event frequencies used in the IPE than in NUREG/CR-4550. In addition, vessel rupture, which contributes a CDF of 9E-8/yr to the IPE results, was only qualitatively assessed and eliminated from further analysis in NUREG/CR-4550.

The licensee performed a human reliability analysis (HRA) and identified that the important operator-related actions are: operator action to depressurize the reactor to the condensate injection pressure, operator action for manual depressurization after inhibiting automatic depressurization (ADS) (with and without offsite power), operator action to recover offsite power, operator action to cross-tie emergency power sources to buses, operator failure to recover feedwater in the short term, operator failure to initiate standby liquid control (SLC) during a main steam isolation valve (MSIV) closure, operator action to recover a diesel generator, operator action to initiate SLC during a turbine trip ATWS with a stuck-open valve, operator action to vent given residual heat removal (RHR) hardware failure, and operator action to inhibit ADS (with high-pressure injection unavailable during ATWS).

To analyze internal flooding, the licensee performed walkdowns; identified flood zones; and developed and quantified flood scenarios based on the flood sources and their impact on critical equipment either by accumulation or direct contact (i.e., sprays or drips). Fifteen flood significant zones were identified with a total flood induced CDF of 1.5E-7/yr. Of these, zones with a CDF above 1E-8/yr are: the reactor building, diesel generator building, turbine buildings, and circulating water pump structure. Internal flooding was screened from the Peach Bottom, Unit 2, NUREG/CR-4550 analysis because of low estimated initiating event frequency.

In accordance with the resolution of Unresolved Safety Issue (USI) A-45, the licensee specifically examined the decay heat removal (DHR) function for vulnerabilities. The licensee used both quantitative design objectives from the NRC staff and qualitative insights from past A-45 studies as input for the analysis of the adequacy of DHR, and concluded that no vulnerabilities associated with DHR exist, since the total CDF from a loss of DHR was below the screening criteria in NUREG-1289. [The submittal's total CDF of 6E-6/reactor-yr includes DHR-failure-related events. NUREG-1289 defines plants with CDF due to DHR failures less than 3E-5/reactor-yr as having "frequency of core damage due to DHR function acceptably small..."]. The staff concludes that the licensee addressed USI A-45 in accordance to GL 88-20.

The back-end portion of the IPE submittal provided the information requested in the IPE Submittal Guidance. Significant PRA findings on the back-end portion of the IPE submittal are discussed here. Cases with no containment failure represented 46% of the CDF. In NUREG/CR-4551 (the NUREG-1150 plant containment analyses), cases with no containment failure were much lower, 18%. The difference appears to be the result of two factors. First, the IPE credited the results of NRC-sponsored work that was completed after NUREG/CR-4551 that indicated a lower probability of drywell liner melt-through than had been used for NUREG/CR-4551. Second, the IPE has a smaller contribution from ATWS sequences, and so a lower percentage of cases with containment failure/venting before core damage.

Consistent with the difference in cases with no containment failure, the IPE submittal reported a lower relative contribution to CDF from early failures (28%), and a higher relative contribution from late failures (26%).

The most frequent release category in the IPE was a low-low release (less than 0.1% CsI release) with an early release time (0 - 6 hrs). The most frequent release category from NUREG/CR-4551 was high release (greater than 1% CsI release) with an intermediate release time (6 - 24 hrs). This difference is consistent with the differences in containment performance discussed above.

After the release categories were further grouped, the highest frequency releases were found to be those with little/no release or low risk impact (versus moderate or high release). The differences observed in the equivalent NUREG/CR-4551 results appear to be due to the different containment performance results discussed above.

In the IPE submittal, the licensee indicated that it has chosen not to evaluate for closure the other USIs and generic safety issues (GSIs) that remain open for Peach Bottom Units 2 & 3 at present.

The IPE addressed the recommendations for plants with Mark I containments from the Containment Performance Improvement (CPI) Program, as summarized below.

Alternate Water Supply for Drywell Spray/Vessel Injection: The submittal indicated that Peach Bottom already has alternative injection/spray capability because it (1) has the capability to inject high-pressure service water through the RHR system, and although the system is ac-dependent, it has the capability of cross-tieing; and (2) it has procedures for using fire water for vessel injection through the RHR system.

Enhanced Reactor Pressure Vessel (RPV) Depressurization System Reliability: The submittal discusses the DC and nitrogen dependencies of the safety relief valves, but the design does not appear exceptional relative to other BWR Mark I plants. No commitments were made toward the CPI objective of enhanced reliability in station blackout.

Emergency Procedures and Training: As recommended by the CPI program, Peach Bottom has implemented Revision 4 of the BWR Owners Group Emergency Procedures Guidelines and is in the process of training the operators.

Torus Hard-Pipe Vent: The licensee installed a hard-pipe vent on both Unit 2, (1992) and Unit 3 (1993). This vent is in addition to the smaller hard-pipe vent from the torus that was modeled in the NUREG-1150 analysis.

The probabilities of containment failure (assuming core damage) given in the IPE are: [with respect to failure location] drywell liner 19 percent, other failure 20 percent, bypass 0.1 percent, vented 14 percent, and intact 46 percent; [with respect to failure time] early 28 percent, late 26 percent, bypass (i.e. instantaneous failure) 0.1 percent, and intact (i.e. no containment failure) 46 percent.

The staff finds the licensee's response to containment performance improvement (CPI) program recommendations consistent with the intent of GL 88-20 and associated Supplement 3.

The licensee identified several unique plant safety features: four diesel generators with the flexibility of cross-tieing buses, residual heat removal and high-pressure service water cross-tieing capability, four 100% capacity residual heat removal pumps and heat exchangers, high-head condensate pumps, MSIV closure at Level 1, and diverse sources of water for cooling and injecting.

The licensee identified the following improvement for implementation: an enhancement to the LOSP procedure which includes detailed instructions to cross-tie emergency electrical buses and recognizes inter-unit interactions to improve the responses necessary for the safe shutdown of Peach Bottom Units 2 & 3 during an LOSP.

## III. CONCLUSION

The staff concludes that, overall, the licensee's treatment of core damage frequencies and radioactive releases is complete with respect to the information requested, and in light of the comparison with NUREG/CR-4550, appears reasonable. Therefore, the Peach Bottom Units 2 & 3 IPE process is acceptable for meeting the intent of GL 88-20. The licensee's intent to continue to use and maintain its IPE/PRA will enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the lifetime of the plant.

Date: October 25, 1995

Attachment: Individual Plant Examination Insight Report for NUREG-1150 Plants, Sandia National Laboratories, August 4, 1995