UNITED STATES
NUCLEAR REGULATORY COMMISSION
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STAFF EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO INDIVIDUAL PLANT EVALUATION (IPE)

PUBLIC SERVICE ELECTRIC & GAS COMPANY

ATLANTIC CITY ELECTRIC COMPANY

HOPE CREEK GENERATING STATION

DOCKET NO. 50-354

I. INTRODUCTION

On May 31, 1994, the Public Service Electric and Gas Company (PSE&G) submitted the Hope Creek Generating Station (HCGS) Individual Plant Examination (IPE) Submittal in response to Generic Letter (GL) 88-20 and associated supplements. On May 12, 1995, the staff sent questions to the licensee requesting additional information. The licensee responded in a letter dated November 6, 1995.

A "Step 1" review of the HCGS IPE submittal was performed and involved the efforts of Science & Engineering Associates, Inc., Concord Associates, and Scientech, Inc./Energy Research, Inc., in the front-end analysis, human reliability analysis (HRA), and back-end analysis, respectively. The Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered (1) the completeness of the information and (2) the reasonableness of the results given the HCGS design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. A summary of contractors' findings is provided below. Details of the contractors' findings are in the attached technical evaluation reports (Enclosure 2, 3, and 4) of this staff evaluation (SE).

In accordance with GL 88-20, PSE&G proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." Generic Safety Issue (GSI) 105, "Intersystem LOCA Outside Containment," was proposed to be resolved as part of the HCGS IPE. No other specific USIs or GSIs were proposed for resolution.

II. EVALUATION

HCGS is a single unit General Electric BWR-4 reactor housed in a Mark I containment. The HCGS IPE has estimated a core damage frequency (CDF) of 4.6E-5/reactor year from internally initiated events, including the contribution from internal floods. The HCGS CDF compares reasonably with that of other BWR-4 plants. Transients with loss of all high pressure injection and failure to depressurize contribute 43% to the overall CDF, station blackout 18%, loss of coolant accidents (LOCAs) 16%, plant-specific initiating

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Enclosure 1

events 11%, and anticipated transients without scram 6%. The important system/equipment contributors to the estimated CDF that appear in the top sequences are: failure of high pressure coolant injection, reactor core isolation cooling, diesel generators, station service water (SSW), and station auxiliary cooling systems (SACS). LOCAs outside of containment were screened from the HCGS IPE. Many IPEs for boiling water reactors (BWRs) have analyzed LOCAs in steam, feedwater, and High Pressure Coolant/Injection/Reactor Core Isolation Cooling (HPCI/RCIC) lines outside containment and found them not to be significant contributors to overall CDF. The licensee's Level 1 analysis appears to have examined the significant initiating events and dominant accident sequences.

Based on the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and review of HCGS plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45 (Decay Heat Removal Reliability) resolution. The licensee evaluated the DHR vulnerabilities related to residual heat removal (RHR), primary coolant system, and containment venting. Loss of DHR was found to contribute only 0.6% to the overall CDF. This low contribution was attributed to redundancy in the RHR system, the ability to cross-tie Diesel Generator (DG) cooling from the two SACS loops and the design characteristics of the SACS system. No other DHR-related vulnerabilities were noted; the licensee considers USI A-45 resolved. The staff agrees. The licensee also provided an analysis related to intersystem LOCA to address GSI 105, "Intersystem LOCA Outside Containment." Results of the analysis indicate that the CDF contribution from Intersystem LOCA is insignificant, specifically 1.7E-9/reactor year or less than 1% of the plant total CDF. The staff agrees that no further licensee action is needed to resolve this issue.

The licensee performed a Human Reliability Analysis (HRA) to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery of failure events. The licensee identified the following operators' actions as important in the estimate of the CDF: failure to recover offsite power, failure to depressurize reactor pressure vessel (RPV), failure to align core spray to the condensate storage tank for long-term injection, and failure to provide alternate ventilation to panel room within 12 hours of loss of heating, ventilation and air conditioning.

The licensee evaluated and quantified the results of the severe accident progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses. The licensee's back-end analysis appeared to have considered important severe accident phenomena. This Level 2 evaluation and quantification were carried out by the licensee for a reduced total CDF value from 4.6E-5/reactor year to 1.3E-5/reactor year. This reduced CDF reflects the credit taken for a modified success criterion for the SACS and SSW systems, which reduced station blackout CDF from 3.4E-5/reactor year to 2.3E-6/reactor year. Based on the original CDF of 4.6E-5, among the HCGS conditional containment failure probabilities, early containment failure was 62% with drywell liner melt-through as the primary contributor, and late containment failure was 28% with late overtemperature as the largest contributor, followed by sump ablation. The identification of sump ablation as a possible failure mode, together with the higher conditional probabilities of late overtemperature failure, lead to the calculation of higher conditional probabilities of late containment failure in the IPE submittal.

The licensee's responses to containment performance improvement (CPI) program recommendations are consistent with the intent of GL 88-20 and associated Supplement 3, although the recommendations of the CPI program have not been directly addressed in the HCGS submittal. The licensee indicated that a hardened vent system has been installed in the HCGS plant. However, the licensee did not consider methods to improve reliability of the Automatic Depressurization System.

Some unique plant safety features identified at HCGS by the licensee are:

1. Hardened Torus Vent - important mitigating feature for loss of decay heat removal sequences.

2. A filtration, recirculation and venting system is located inside the reactor building under negative pressure and is capable of filtering fission products from the building once a release into the building occurs.

3. Alternate water supply for drywell spray/vessel injection.

4. Alternate sources of injection to the RPV (firewater and service water) were identified and connection lines to the RPV installed.

The licensee stated that for a sequence or an event to be considered indicative of a vulnerability, it had to pass the screening criteria for reporting systematic sequences from NUREG-1335 and contribute inordinately to the CDF with respect to either other sequences or events in the IPE, or in comparison with PRA results for other plants. During the performance of the IPE, transients involving heating, ventilation and air conditioning (HVAC) failures were determined to contribute inordinately to the CDF. For example, loss of switchgear or 1E panel room HVAC had a CDF of 3.3E-3/reactor year. This loss was labeled as a vulnerability, and a procedure to provide alternate ventilation was developed. This procedure requires operator action to open the door of the switchgear room and to provide a portable fan for alternate ventilation. Based on this procedure, the licensee performed an analysis by the use of Electric Power Research Institute methodology to account for potential human errors and the results of the analysis indicate that implementation of the procedure reduced the CDF for the loss of HVAC sequences from 3.3E-3/reactor year to 1E-6/reactor year. Therefore, the implementation of the procedure removed this vulnerability. While the staff notes the licensee's efforts to address this vulnerability, the staff believes the reduction of this sequence by three orders of magnitude may be overly optimistic. It is not clear from the submittal if the licensee considered in the analysis such items as availability of portable fans, ability to power fans during any and all applicable sequences, indication of switchgear room heatup and available versus needed time, operator training (since new procedure, unfamiliarity exists). It is recommended that the licensee should re-examine the analysis to verify these calculated CDFs.

No other vulnerabilities were identified by the licensee. In addition, no other improvements were proposed by the licensee.

A qualitative criterion was used to determine vulnerabilities related to containment perfo mance. A vulnerability was identified if the HCGS containment performance results were significantly different from similar BWRs. The licensee stated that the relatively large frequencies and conditional probabilities of large-early and medium-early releases observed by this comparison were due to the dominant contribution of the station blackout sequences to the CDF. However, by crediting the modified success criterion for the SACS and SSW systems, the station blackout CDF was reduced. Thus, the frequency of early and late containment failure, and early-high and early-medium releases, are all expected to decline. No hardware modifications, based on this back-end analysis, have been planned.

III. CONCLUSION

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regards to the information requested by GL 88-20 (and associated guidance NUREG-1335), and (2) the IPE results are reasonable given the HCGS design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the HCGS IPE has met the intent of GL 88-20.

It should be noted, that the staff's review primarily focused on the licensee's ability to examine HCGS for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SE does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20.

HOPE CREEK GENERATING STATION

INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)

Enclosure 2