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

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2,

INDIVIDUAL PLANT EXAMINATION

STAFF EVALUATION REPORT

1. INTRODUCTION

On August 31, 1993, the Carolina Power and Light Company submitted the Brunswick Steam Electric Plant, Units 1 and 2, Individual Plant Examination (IPE) in response to Generic Letter (GL) 88-20 , "Individual Plant Examinations for Severe Accident Vulnerabilities," and associated supplements. The licensee supplemented the IPE by letters dated September 9, 1994, September 30, 1994, February 27, 1995, and May 18, 1995.

A "Step 1" review of the Brunswick IPE submittal was performed and involved the efforts of Science and 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 Brunswick design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. A summary of the contractors' findings is provided below. Details of the contractors' findings are in the attached technical evaluation reports (Enclosures 2, 3, and 4) of this staff evaluation report (SER).

In accordance with GL 88-20, CP&L proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." USI A-17, "Systems Interactions in Nuclear Power Plants," was also proposed to be resolved as part of the Brunswick IPE. No other specific USIs or generic safety issues were proposed for resolution as part of the Brunswick IPE.

2. EVALUATION

Brunswick is a two-unit General Electric BWR-4 reactor design with each reactor housed in a Mark I containment. The Brunswick containments are steel-lined with concrete which are different from the typical Mark I steel shell designs used for most BWR-4s, such as those in the Browns Ferry and Peach Bottom plants. The Brunswick IPE has estimated a core damage frequency (CDF) of 2.7E-5 per reactor-year from internally initiated events, including the contribution from internal floods. The Brunswick CDF compares reasonably with that of other BWR-4 plants. Loss-of-offsite power transients including station blackout contribute 66%, transients with loss of decay heat removal contribute 30%, and anticipated transients without scram (ATWS) contribute 3%. The important system/equipment contributors to the estimated CDF that appear in the top sequences are: failures of the diesel generators, instrument air system, residual heat removal system, and service water system. 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 Brunswick plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45 resolution. The licensee proposed to resolve USI A-17 by the Internal Flooding analysis. Although the flooding analysis resulted in flood-induced core damage sequences, they were not dominant contributors to the overall CDF. The staff finds USI A-17 resolved for Brunswick since no vulnerabilities from internal flooding were identified.

The licensee performed an 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 operator actions as important in the estimate of the CDF: failure to recover offsite power within a few hours; failure to use the safety relief valves to depressurize; failure to correctly initiate suppression pool cooling; failure to inhibit the automatic depressurization system valves during an anticipated transient without scram (ATWS); failure to vent or to control venting; and failure to actuate the standby liquid control system.

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 was carried out by the licensee for a reduced total CDF value of 1.9E-5 per reactor year. This reduced CDF reflects the credit taken in the Level 2 thermal hydraulic analyses to account for the availability of the control rod drive system for accident sequences involving loss of decay heat removal and ATWS. Based on the reduced total CDF value of 1.9E-5 per reactor year, among the Brunswick conditional containment failure probabilities, early containment failure is about 12% with overpressure and vessel thrust forces as the primary contributors, late containment failure is about 85% with overpressure being the primary contributor, and the bypass is less than 1% with interfacing systems loss-of-coolant accidents the primary contributor. Containment venting after core damage is assumed to occur in about 1% of the sequences. In addition, the containment remained intact with a breached reactor vessel about 1% of the time.

The licensee evaluated the "Mark I Containment Performance Improvements (CPI)" discussed in GL 88-20, Supplement 1. Subsequent to the August 1992 IPE submittal (with a freeze date of January 1, 1992), CP&L installed a hardened wetwell vent in both Brunswick units in response to GL 89-16, "Installation of a Hardened Wetwell Vent." The August 1992 IPE did not take credit for this modification. The supplemental letters informed us that in the updated PRA, the hardened vent reduced the total CDF by approximately 10%. CP&L had adopted Revision 4 of the BWR Owners Group Emergency Procedure Guidelines (BWROG EPGs). The IPE took credit for the revised emergency operating procedures (EOPs) and augmented operator training. With respect to the use of the firewater system as an alternate water supply for drywell spray/vessel injection recommended by the CPI program, the licensee indicates that the firewater system for Brunswick will result in minimum benefit. The licensee's analysis shows that the potential use of the firewater system would be in accident sequences involving station blackout, which contribute about 70 percent of the plant total CDF. During these accidents, the drywell and vessel pressure will maintain at a relatively high value in comparison with the shutoff head of the firewater pump. Consequently, the flow of the firewater system is reduced, and the potential fission product scrubbing by the firewater becomes ineffective. Further, during certain periods of the transients, the drywell and vessel will repressurize. As such, firewater injection into the drywell or vessel will be precluded. Based on this analysis, the licensee is not committed to the use of the firewater system as an alternative water source for drywell spray/vessel injection. CP&L adequately addressed all of the NRC-suggested CPI items.

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

Some unique plant safety features identified at Brunswick are:

1. Ability to cross-tie the 1E buses between units.

2. Ability to vent containment using the containment atmospheric control system and the standby gas treatment system.

3. Ability to flood the core and containment with service water pumps or the diesel-driven fire pump via the RHR system.

The licensee used the criteria in Nuclear Management and Resources Council (NUMARC) 91-04, "Severe Accident Issue Closure Guidelines," to screen for plant-specific vulnerabilities. The licensee did not identify any vulnerabilities, but determined that some modifications were warranted to address weaknesses identified by the IPE. These improvements, listed below, were not credited in the August 31, 1992, IPE submittal but were reflected in the reduced CDF reported in the supplemental letters of September 1994 and February 1995.

1. Installation of a remotely operated emergency bus cross-tie and logic switches to cross-tie the 4160-V buses between Units 1 and 2.

2. Development of new load shedding procedures that more than double the time to battery depletion.

3. Installation of a hardened wetwell vent.

4. Upgrading the capacity of the secondary Y-winding non-segregated bus of the startup auxiliary transformers (SATs) from 4000 to 5000 amps.

5. Installation of no-load disconnect switches and logic that provide the capability of restoring offsite power in less than an hour via a backfeed from the switchyard through the main and unit auxiliary transformers.

6. Development of the necessary procedures for operation of the above hardware improvements.

3. CONCLUSION

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance in NUREG-1335); and (2) the IPE results are reasonable given the Brunswick 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 Brunswick 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 Brunswick 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 SER 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.