NRC STAFF EVALUATION REPORT

OF THE INDIVIDUAL PLANT EXAMINATION FOR THE COOPER NUCLEAR STATION

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

I. INTRODUCTION

On March 31, 1993, the Nebraska Public Power District (NPPD) submitted the Cooper Nuclear Station (CNS) Individual Plant Examination (IPE) in response to Generic Letter (GL) 88-20 and associated supplements. On October 21, 1994, the staff sent questions to the licensee requesting additional information. The licensee responded in a letter dated February 20, 1995.

A "Step 1" review of the CNS IPE submittal was performed and involved the efforts of Science & Engineering Associates, Inc., Scientech, Inc./Energy Research, Inc., and Concord Associates in the front-end, back-end, and human reliability analysis (HRA), 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 IPE 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 technical evaluation reports (Enclosures 2, 3, and 4 of the letter).

In accordance with GL 88-20, CNS proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." No other specific USIs or GSIs were proposed for resolution as part of the CNS IPE.

II. EVALUATION

CNS is a single unit BWR 4 with a Mark I containment. The CNS IPE has estimated a core damage frequency (CDF) of 8E-5/reactor-year from internally initiated events, excluding internal floods. The licensee did not quantify the CDF from internal flooding. NPPD relied on a previous study of CNS performed by Sandia National Laboratories (NUREG/CR-4767, "Shutdown Decay Heat Removal Analysis of a General Electric BWR 4/Mark I"), which concluded that no areas (e.g., switchgear rooms, pump rooms, etc.) at CNS were vulnerable to internal flood-induced core damage accidents with a frequency greater than 1E-6/reactor year.

Enclosure 1

The CNS CDF is the highest CDF calculated by any licensee for a BWR. The high CDF is the result of a number of factors, such as suppression pool lost at 200°F as a source of water for core cooling, no credit for hardened vent, no credit for load shedding. Station blackout contributes 35%, transient induced loss of coolant accidents (LOCAs) 30%, loss of coolant injection 18%, loss of containment heat removal 11%, anticipated transients without scram 5%, and LOCAs 1%. The important system/equipment contributors to the estimated CDF that appear in the top sequences are: diesel generators (DGs), high pressure coolant injection system (HPCI), reactor core isolation cooling (RCIC) system, common cause failure (CCF) of standby service water pumps, and CCF of the safety relief valves (SRVs). 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 CNS 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 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 operator actions as important in the estimate of the CDF: operator inhibits automatic depressurization system, recovery of loss of offsite power, depressurize with the SRVs, recovery of the DGs, recovery of service water from repair, restore reactor pressure switches, recovery of critical switchgear room ventilation within one hour, recovery of DG CCF, recovery of DG from maintenance, and prevent rapid overfill with HPCI unavailable.

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. Among the CNS conditional containment failure probabilities early containment failure is 36% with drywell liner melt-through as the primary contributor and late containment failures is 31%. The containment remains intact (including venting through 2" line via the standby gas treatment system) 33% of the time. Early radiological releases are dominated by liner melt-through (99% of early releases). The licensee's response to containment performance improvement program recommendations is consistent with the intent of Generic Letter 88-20 and associated Supplement 3; however, CNS is evaluating the potential benefit of two of the recommendations (alternate water supply for the drywell sprays and enhancing the reactor pressure vessel depressurization) as part of their upgrade.

Some insights and unique plant safety features identified at CNS are:

- Consideration of upgraded emergency operating procedures.

- Load study to relax the assumed four-hour battery lifetime.

- Improved reliability data (obtained after the freeze date) for HPCI and RCIC.

- Nitrogen supply to the SRVs depends on AC power, a procedure to bypass the AC solenoid valve would reduce risks by ensuring a pneumatic supply to the SRVs.

- A diesel driven fire water pump or other similar source of low pressure water independent of AC power would provide a significant reduction in risk for station blackout sequences.

- Capability of the reactor building to retain some of the radionuclides released from the primary containment.

- Early failure of the drywell head seal, which would result in the retention of a significant fraction of radionuclides in the containment, and the use of containment vents to reduce containment pressure and prevent large failure of the containment, thus reducing the amount of radionuclides released.

- Presence of a copious amount of water on the drywell floor to quench corium after vessel breach, which would greatly reduce the probability of drywell liner melt-through.

- Capability of the drywell sprays to effectively reduce pressure and temperature once the sprays were recovered during the accident.

The licensee's definition of vulnerability was based on the criteria provided in Nuclear Management and Resources Council (NUMARC) 91-4, "Severe Accident Issue Closure Guidelines," January 1992, and varied on the basis of relative aspects of core damage frequency, containment bypass, and release significance. Based on this criteria, the licensee did not identify any vulnerabilities and stated that "no modifications are planned based on these insights." NPPD did indicate, however, that "when the model is upgraded to reflect the latest revisions of the EOPs and other modifications made since the cutoff date, these insights will be revisited." The licensee did note the following improvements (separate from the IPE analysis):

- Extensive modifications to environmentally qualify components to continue to function in harsh environments under accident conditions.

- New EOPs that permit earlier spraying of the containment, thus providing the water necessary to mitigate some of the drywell liner melt-through accidents.

- Hardened vent system.

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 in NUREG-1335), and (2) the IPE results are reasonable given the CNS 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 CNS 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 CNS 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 staff evaluation report 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.