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

RELATED TO THE INDIVIDUAL PLANT EVALUATION (IPE)

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

On December 1, 1993, Omaha Public Power District submitted the Fort Calhoun Nuclear Power Plant Individual Plant Evaluation (IPE) submittal in response to Generic Letter 88-20 and associated supplements. On September 12, 1995, the staff sent questions to the licensee requesting additional information. The licensee responded in a letter dated November 30, 1995.

A "Step 1" review of the Fort Calhoun 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 Fort Calhoun design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. Details of the contractors' findings are in the attached technical evaluation reports (Appendices A, B, and C) of this staff evaluation report (SER).

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

The submittal states that the licensee intends to maintain a "living" probabilistic risk assessment.

2.0 EVALUATION

Fort Calhoun is a Combustion Engineering PWR with a large containment. The Fort Calhoun IPE has estimated a core damage frequency (CDF) of 1.4E-05 per reactor-year from internally initiated events, including the contribution of 2E-06 from internal floods. The Fort Calhoun CDF compares reasonably well with that of other PWR plants. Station blackout contributes 35 percent; transients, 31 percent; internal flooding, 14 percent; loss of coolant accidents (LOCAs), 8 percent; steam generator tube rupture, 6 percent;

interfacing systems LOCA (ISLOCA), 5 percent; and anticipated transients without scram, 2 percent. The most important system/equipment contributors to the estimated CDF that appear in the top sequences are:

1. Common cause unsuccessful load shed from 4.16 kV AC buses 1A3 and 1A4.

2. Failure of the diesel driven auxiliary feedwater pump.

3. Failure of RCP seals given insufficient cooling.

4. Diesel generator failure to run.

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 Fort Calhoun plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the USI A-45 (DHR Reliability) resolution and is, therefore, acceptable.

The licensee performed an HRA, including both pre- and post-initiator human actions 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, based on the Fussel-Vesely importance measure, in the estimate of the CDF:

1. Operator failure to use diesel driven feedwater pump to replenish emergency feedwater storage tank.

2. Operator fails to use diesel driven fire pump to replenish emergency feedwater storage tank.

3. Operator fails to manually trip 4.16 kV AC circuit breaker, given that breaker does not trip automatically.

However, there appear to be certain limitations in the analysis. For example, there are some characteristics associated with the modeling of mistakes that can lead to seemingly inconsistent results. The model uses different time/reliability correlations depending on whether actions are verification, rule-based, or "other" actions, whether they occur inside or outside the control room, and whether the operators are burdened. It appears that differences in quantification results based on these correlations may be significant. In addition, there appears to be no specific guidance as to which actions should be assigned to include burden. This factor, also, can affect the estimated failure probability.

Despite these limitations, the Fort Calhoun HRA analysis appears to include all the appropriate classes of human actions that are likely to contribute to the frequency of core damage, such as, maintenance, test and calibration actions in the pre-accident phase, and failure in decision-making (mistakes) and task execution (slips) in the post-accident phase. It also explicitly

describes how the human actions should be incorporated into the PRA logic models. For these reasons, the staff believes the HRA portion of the analysis, while containing the weaknesses discussed above, does consitute an adequate component of the IPE analysis in the search for vulnerabilities.

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. According to the licensee, the Fort Calhoun conditional containment failure probabilities are as follows: early containment failure (defined as that occurring at or within one hour of reactor vessel failure), two percent with hydrogen combustion and direct containment heating the primary contributors; late containment failures, 28 percent with overpressure failure caused by loss of containment heat removal being the primary contributor; bypass five percent with ISLOCA being the primary contributor; and containment isolation failure, five percent, with SGTR (along with assumed failure to isolate the affected steam generator) being the primary contributor. Alpha mode failure and basemat melt-through failures were reported to be negligible. According to the licensee, the containment remains intact 60 percent of the time. Early radiological releases are dominated by ISLOCA and SGTR and late releases are dominated by station blackout and sequences. The licensee's response to containment performance improvement program recommendations is consistent with the intent of Generic Letter 88-20 and associated Supplement 3.

According to the licensee, some insights and unique plant safety features identified by the licensee at Fort Calhoun are:

1. Ability to feed and bleed once through cooling.

2. Use of self contained radiators for diesel generator cooling which do not require external cooling from plant cooling water systems.

3. Diverse means of supplying AFW to the steam generators, i.e., by either a motor driven, turbine driven, or diesel driven pump. Without credit for the diesel driven AFW pump, the CDF would increase by a factor of 5.

4. More robust design (according to the licensee) of the reactor coolant pumps (RCPs) which are stated to be highly resistant to seal leakage. Without this assumed enhanced performance the CDF would increase by a factor of 10.

5. Lack of a requirement for emergency core cooling system (ECCS) pump external cooling during the injection mode.

6. Lack of a "piggy-back" requirement for high pressure coolant injection (HPCI) pumps from low pressure coolant injection (LPCI) pumps during recirculation.

7. It was assumed that a large LOCA could be mitigated without the use of LPSI pumps; specifically during the early phase of a large LOCA, one HPSI pump and three safety injection pumps meet the success criteria. This success criteria is more optimistic than reported in many PWR IPE submittals which typically assume that the large LOCA requires at least one LPSI pump.

7. Automatic switchover of ECCS from injection to recirculation.

8. Open design of the auxiliary building, which encourages natural circulation, making it unlikely that heating, ventilation, and air conditioning (HVAC) will be required to cool many items of plant equipment.

9. The plant design includes a containment air cooling and filtering system, which provides containment cooling independent of the containment spray system.

10. Ability to use a diesel driven fire pump for plant functions, such as, for delivering long term makeup to the emergency feedwater storage tank and for providing backup cooling to the component cooling water system.

The licensee adopted criteria from the Nuclear Management and Resource Council (NUMARC) to screen for plant-specific vulnerabilities. These criteria were applied to the functional core damage sequences. Based on this definition, the licensee did not identify any vulnerabilities. Plant improvements, however, were identified. These improvements, listed below, have been implemented, with the exception of four (4), which is still in progress:

1. Install a door to facilitate mitigation of RCP seal cooler ISLOCA.

2. Periodically leak test downstream shutdown cooling valve (on ISLOCA path.)

3. Install anti-galloping devices on 161 kV offsite power source.

4. For internal flood scenarios, revise procedures to establish appropriate position of the door to the spent/regenerative tank/pump room.

Taken together, the licensee reported that the total CDF reduction from the four improvements listed above was 1.8E-05, which then resulted in the reported actual CDF of 1.4E-05.

3.0 <u>CONCLUSION</u>

Based on the above findings, the staff notes that: (1) the licensee's IPE is complete with regard to the information requested by Generic Letter 88-20 (and associated guidance in NUREG-1335), and (2) the IPE results are reasonable given the Fort Calhoun 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 Fort Calhoun IPE has met the intent of Generic Letter 88-20.

It should be noted that the staff's review primarily focused on the licensee's ability to examine Fort Calhoun 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 Generic Letter 88-20. The staff has identified a weakness in the HRA portion of the IPE and believes that application of the IPE in support of risk-based regulatory applications, beyond those associated with Generic Letter 88-20, require additional treatment in that area.

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APPENDIX A

FRONT-END TECHNICAL EVALUATION REPORT