50-461

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

March 27, 1997

Mr. Paul J. Telthorst
Director - Licensing
Clinton Power Station
P. O. Box 678
Mail Code V920
Clinton, IL 61727

SUBJECT: STAFF EVALUATION OF CLINTON POWER STATION INDIVIDUAL PLANT EXAMINATION - INTERNAL EVENTS (TAC NO. M74396)

Dear Mr. Telthorst:

Enclosed is the staff's evaluation for the Clinton Power Station individual plant examination (IPE) submittal for internal events and internal flooding. The contractor's technical evaluation reports (TERs) are included with the evaluation.

The staff performed a "Step 1" review during which it examined the IPE results for their "reasonableness," taking into consideration the design and operation of Clinton Power Station. Science & Engineering Associates, Inc., Concord Associates, and Scientech, Inc. reviewed the front-end analysis, human reliability analysis, and back-end analysis, respectively, of the IPE submittal. Their TERs are enclosed as Appendices A, B, and C, respectively, to the staff's evaluation. These TERs were reviewed by the IPE Senior Review Board (SRB) as part of the Office of Nuclear Regulatory Research (RES) quality assurance process. The SRB consists of RES staff and consultants at Sandia and Brookhaven National Laboratories with probabilistic risk assessment (PRA) expertise.

In the IPE, you estimated a total core damage frequency (CDF) of about 3E-5/reactor-year, including a contribution from internal flooding of about 2E-6/reactor-year. Transients contribute 52 percent, station blackout 37 percent, internal flooding 6 percent, loss-of-coolant accidents 4 percent, and anticipated transient without scram 1 percent.

To identify vulnerabilities, you addressed the following questions: (1) Are there new or unusual means by which core damage or containment failure could occur compared to those identified in other PRAs; (2) Do the results suggest that the plant CDF would not be able to meet the NRC's subsidiary safety goal for core damage (1E-04/reactor-year); and (3) Are there any systems, components, or operator actions that control the core damage result (i.e., greater than 90 percent)? Your examination did not lead to the identification of any vulnerabilities. Several plant improvements, however, were identified and implemented.

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On the basis of the "Step 1" review, we conclude that you have met the intent of Generic Letter 88-20. The staff does not recommend that a more detailed, "Step 2" review be conducted. We note, however, that identified weaknesses in the IPE may limit its use for any other regulatory purposes.

If you have any questions regarding the enclosed evaluation, please feel free to contact me at (301) 415-1364.

Sincerely,

Original signed by:

Douglas V. Pickett, Senior Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-461

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CLINTON NUCLEAR POWER PLANT INDIVIDUAL PLANT EXAMINATION

STAFF EVALUATION REPORT

ENCLOSURE

I. INTRODUCTION

On September 23, 1992, Illinois Power (licensee) submitted the Clinton Nuclear Power Plant individual plant examination (IPE) in response to Generic Letter (GL) 88-20 and associated supplements. On July 21, 1995, the staff sent a request for additional information to the licensee. The licensee responded in a letter dated November 22, 1995.

The staff performed a "Step 1" review of the Clinton IPE submittal. As part of this review, Science & Engineering Associates, Inc., Scientech, Inc., and Concord Associates reviewed the front-end analysis, back-end analysis, and the 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 Clinton design, operation, and history. A more detailed review, a "Step 2" review, was not performed as part of this IPE submittal. Details of the contractors' findings are given in the technical evaluation reports (Appendices A, B, and C) attached to this staff evaluation report (SER).

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

II. EVALUATION

Clinton Nuclear Power Station is a boiling water reactor (BWR) 6 with a Mark III containment. The licensee estimated a core damage frequency (CDF) of about 3E-5/reactor-year from internally initiated events, including a contribution from internal flooding of about 2E-6/reactor year. The Clinton CDF compares reasonably with that of other BWR 6 plants. Transients contribute 52 percent, station blackout 37 percent, internal flooding 6 percent, loss-of-coolant accidents 4 percent, and anticipated transient without scram 1 percent.

On the basis of the licensee's Fussell-Vesely importance analysis, the most important contributors to the estimated CDF sequences are associated with loss of offsite power, failure of the high pressure core spray (HPCS) system, failure of the reactor core isolation coolant (RCIC) system, transients without isolation, transients with isolation, failure of the fire protection system, and failure of the automatic depessurization system (ADS).

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 in 0.5 hour, failure to recover HPCS, failure to recover RCIC, failure to manually initiate ADS, failure to recover Division 2 within the first 4 hours, and failure to manually initiate service water.

The licensee's Level 1 analysis appears to have examined the significant initiating events and dominant accident sequences. However, it does not appear that the licensee incorporated plant-specific experience in the calculation of equipment failure rates and initiating event frequencies. Responding to staff's RAI, the licensee stated that very limited plant-specific data exists because Clinton is a relatively new plant. The licensee cited examples showing that plant-specific failure data are comparable to or better than the generic data used. However, plant-specific data could have been used to update the generic data by means of a Bayesian process. Insufficient incorporation of plant experience may not have a significant impact on the total CDF but it may have a significant impact on the relative contributions of the various sequences and failure events and, therefore, it may limit the ability to identify plant-specific insights and improvements.

Also, the Clinton IPE credited local repair of various equipment components and systems (including diesel generators, pumps, valves, and instrumentation) and has taken credit for up-to-two component/system repair activities. The licensee stated that the repair activities credited in the Clinton IPE were based on actual emergency exercises; however, the quantification of these repair activities is based on a generic Electric Power Research Institute database.

The licensee, as part of its response to staff's RAI, performed a sensitivity analysis that involved removal of all credited equipment repair recoveries, except those involving offsite power, direct current power, and operator actions performed from the control room. The result was an insignificant increase of the CDF from about 3E-05/reactor-year to about 4E-05/reactor-year, a factor of 1.5. The relative contributions of individual accident sequences also were not significantly altered. In no instances were increases in individual accident sequence frequencies greater than a factor of 2.4. Although two new sequences were introduced, their frequencies were less than 1E-08/reactor-year.

The success of equipment repair depends on many important plant-specific factors such as the type of failure, time needed for diagnosis, time needed for repair (which may range from a very few hours to several days), crew competing tasks under different accident conditions, and crew availability (especially when multiple-repairs are credited). These factors do not appear to have been taken into consideration in the Clinton IPE in modeling and estimating equipment repair probabilities under accident conditions. Therefore, although the staff concludes that the credit taken for repair of equipment did not have a significant effect on the IPE's results, the staff believes that insufficient examination of important aspects of human performance under severe accident conditions may have limited the ability to identify factors critical to the accomplishment of the actions modeled.

The Clinton IPE results are in line with those for similar BWR 6 plants in both the most dominant initiators and the most dominant accident sequences. Therefore, the staff believes it is unlikely that the limitations identified have impacted the licensee's overall conclusions from the IPE and its capability to identify vulnerabilities. However, they may have limited its ability to gain insights and identify improvements.

On the basis of the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and review of Clinton plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the resolution of USI A-45 "Shutdown Decay Heat Removal Requirements."

The licensee evaluated and quantified the results of the severe accident progression. The back-end analysis results are as follows: early containment failure (including isolation failure) will occur about 3 percent of the time, late containment failures will occur about 2 percent of the time, and bypass will occur less than 1 percent of the time. The containment remains intact 95 percent of the time.

The staff has also identified a weakness in the Clinton IPE back-end analysis. The licensee, responding to staff's request for additional information, did not provide a justification for the very small probability of containment failure from hydrogen ignition after power recovery under station blackout conditions, or for no containment failure from excessive fuel-coolant interaction. Although, as stated in the IPE, Clinton has "the largest free air volume and suppression pool volume to rated thermal power of any domestic Mark III," it is not clear whether the result that the containment remains intact 95 percent of the time is skewed because of the licensee's optimistic approach to the examination of hydrogen combustion and ex-vessel steam explosion phenomena. It is noted, however, that although certain phenomena may have been treated optimistically, the licensee has considered all relevant phenomena. The licensee developed containment event trees to characterize containment response to severe accidents and examined several severe accident phenomena in detail for their applicability to Clinton. Also, the licensee used the Modular Analysis Accident Program to analyze representative sequences and addressed phenomenological uncertainties of accident progression through sensitivity studies. These findings coupled with the fact that the Clinton containment is substantially stronger than any other Mark III containment leads the staff to conclude that the licensee did not miss a vulnerability with respect Clinton's containment.

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

Some insights and unique plant safety features identified by the licensee are the following:

1. four-hour battery lifetime

2. motor-driven feedwater pump

3. three safety-related divisions of core cooling, each of which has its own emergency diesel generator and cooling water

4. ability of the emergency core cooling system pumps to operate when the suppression pool is saturated
5. ability to cross-connect the fire protection system for core injection

6. a containment with the largest free air volume and suppression pool volume with respect to rated thermal power of any domestic Mark III.

To identify vulnerabilities the licensee addressed the following questions: (1) Are there new or unusual means by which core damage or containment failure could occur compared to those identified in other PRAs? (2) Do the results suggest that the plant CDF would not be able to meet the NRC's subsidiary safety goal for core damage (1E-04/reactor-year)? (3) Are there any systems, components, or operator actions that control the core damage result (i.e., greater than 90 percent). The licensee's examination did not lead to the identification of any vulnerabilities. Several plant improvements, however, were identified. The following improvements have been implemented:

1. operator training to emphasize the importance of maintaining offsite power

2. operator training to emphasize the importance of manual ADS initiation

3. modification of the HPCS surveillance procedure to test the suppression pool suction path

4. installation of a bypass line to allow easier use of the fire protection system for vessel makeup

5. operator training to emphasize the importance of maintaining offsite power to prevent offsite releases

6. operator training to emphasize the importance of alternate current power recovery

7. operator training to emphasize the importance of manually isolating the containment bypass path into the fuel pool cooling/cleanup line during station blackout

8. operator training to emphasize the significance of scram system hardware failures as related to release frequencies

III. CONCLUSION

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 Clinton Nuclear Power Station 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 Clinton Nuclear Power Plant IPE has met the intent of Generic Letter 88-20. However, the staff noted limitations in the licensee's IPE associated with (1) the extensive use of generic data, (2) the credit taken for equipment repair, (3) the credit taken for containment performance under hydrogen combustion upon power recovery (negligible failure probability), and (4) the credit taken for containment

performance under ex-vessel steam-explosion conditions (no containment failure). These limitations may limit the IPE's usefulness for other regulatory applications.

It should be noted that the staff's review primarily focused on the licensee's ability to examine Clinton Nuclear Power Station 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.

APPENDIX A

FRONT-END TECHNICAL EVALUATION REPORT