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

50-482

November 18, 1996

Mr. Neil S. Carns
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

SUBJECT: STAFF EVALUATION REPORT FOR THE REVIEW OF THE WOLF CREEK GENERATING STATION INDIVIDUAL PLANT EVALUATION (TAC. NO. M74490)

Dear Mr. Carns:

Enclosed is the NRC Staff Evaluation Report (SER) for the Wolf Creek Generating Station Individual Plant Evaluation (IPE) for internal events and internal floods. Also included with the SER are the contractors' (Science & Engineering Associates, Inc., Concord Associates, and Scientech Inc.) Technical Evaluation Reports (TERs).

During the review the staff identified two concerns:

(1) A limited set (5) of Human Reliability Analysis (HRA) of calibration actions, including the refueling water storage tank level, which other IPEs have identified as a potentially significant event. However, the basis as to why these were the only events identified for analysis was not provided.

(2) The modeling of errors associated with actions that have to be performed within a very short time (e.g., times in the range of seconds to 1 minute).

The staff does not believe that these possible shortcomings would have prevented the licensee from identifying a vulnerability. The licensee is encouraged, in future revisions of the Wolf Creek IPE, to better document the process used to identify and select pre-initiator (miscalibration) events and to better treat time in modeling of errors associated with actions that have to be performed within a very short time.

Based on the findings discussed in the enclosed reports, the staff concludes that your IPE is complete with regard to the information requested in Generic Letter 88-20 (GL 88-20), and associated guidance in NUREG-1335, and the IPE results are reasonable given Wolf Creek's design, operation and history. As a result, the staff concludes that your IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, the Wolf Creek IPE has met the intent of GL 88-20.

It should be noted that the staff's review focused primarily on your ability to examine the Wolf Creek plant 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 your detailed findings (or quantification estimates) that stemmed from the examination. Therefore, the

enclosed 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.

If you have any questions, please contact me at (301) 415-3063.

Sincerely,

James C. Stone

James C. Stone, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure: Staff Evaluation Report
w/attachments

cc w/encl: See next page

cc w/encl:

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ATTACHMENT 1

WOLF CREEK GENERATING STATION INDIVIDUAL PLANT EXAMINATION

STAFF EVALUATION REPORT

I. INTRODUCTION

On September 28, 1992, the Wolf Creek Nuclear Operating Corporation (WCNOC) provided the Wolf Creek Generating Station (WCGS) Individual Plant Examination (IPE) submittal in response to Generic Letter (GL) 88-20 and associated supplements. On June 28, 1995, the staff sent questions to the licensee requesting additional information (RAI). The licensee responded in a letter dated August 30, 1995. In response to the RAI's and a teleconferences on April 17, 1996, the licensee submitted a modified analysis on May 30, 1996, which discussed the revised human reliability analysis (HRA) and the revised common cause failure (CCF) analysis. Additional information regarding human reliability and common cause failure analyses was submitted to the staff on September 13, 1996 for clarification. The modified analysis also included the impact on core damage frequency (CDF) of these revised analyses and the impact of the conversion of the WCGS probabilistic safety assessment model from the Westinghouse codes originally used for quantification, to the NUS NUPRA code used for quantification in the revised analyses.

A "Step 1" review of the WCGS 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 WCGS 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 GL 88-20, WCNOC proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," and USI A-17 "Systems Interactions." No other specific USIs or generic safety issues were proposed for resolution as part of the WCGS IPE.

II. EVALUATION

WCGS is a Westinghouse 4 loop pressurized water reactor (PWR) with a large dry containment. In its 1992 IPE submittal, the licensee estimated the total CDF for the WCGS as 4.2E-5/reactor-year(ry) for internally initiated events, including internal flooding. The WCGS CDF compares reasonably with that of other Westinghouse plants. Station blackout contributes 45%, internal floods 18%, transients (including loss of offsite power) 13%, special initiators 12% (loss of service water 6%, loss of operating train of component cooling water leading to seal LOCA 5%, total loss of component cooling water 1%), loss of coolant accidents (LOCA) 10%, steam generator tube rupture (SGTR) 1%, and anticipated transients without scram (ATWS) and interfacing systems LOCA <1%. The total CDF estimated by the licensee in its modified analysis is 6.2E-5/ry with station blackout contributing 45%, special initiators 16% (loss of service water 8%, total loss of component cooling water 7%, loss of operating train of component cooling water leading to RCP seal cooling 1%), LOCAs 16%, transients (including loss of offsite power) 12%, ATWS 5%, internal floods 4%, and SGTR 1%.

As noted above, the quantification of the WCGS model using the revised HRA and CCF values resulted in a combined increase in CDF of 47% over the original CDF of 4.2E-5/ry. The licensee indicated that the increase in CDF due to the revised CCF analysis was 34% over the frequency of 4.2E-5/ry identified for the total core damage frequency in the original analysis. In addressing the staff's concerns regarding the low Multiple Greek Letter (MGL) parameters used in the original CCF analysis, the licensee requantified using MGL parameters from EPRI NP-3967 and NUREG/CR-4780. These values are comparable to the values found in NUREG/CR-4550. The revised CCF analysis resulted in increases in contributions from sequences involving SBO (emergency diesel generators and related support systems), large LOCA (recirculation system isolation valves, RHR pumps) and ATWS (reactor trip breakers) and contributed to changes in the ranking of the contributions from these initiating events as noted above.

In addressing the staff's concerns on the HRA, the licensee revised the IPE's HRA significantly. In the revised HRA the licensee searched for pre-initiator human events and included events related to miscalibration (excluded in the original analysis) in addition to the re-alignment of valves after test or maintenance that were addressed in the original HRA. The licensee identified and performed a HRA for a limited set (5) of calibration actions, including the refueling water storage tank level, which other IPEs have identified as a potentially significant event. However, the licensee did not provide a basis as to why these were the only events identified for analysis.

The licensee also completely revised the post-initiator human event analysis. In the revision, the licensee primarily used the "EPRI Cause Based Decision Tree Methodology (CBDTM)" described in EPRI TR-100259, while a "modified THERP" was used in the original analysis. Therefore, the re-analysis provided has substantially eliminated most of staff's concerns associated with the way the "modified THERP" had been applied in the original analysis. The staff finds that the licensee adequately addressed the decision making element of the post-initiator actions and dependencies between human errors, aspects of the analysis that were significant weaknesses in the original analysis. Also the re-analysis eliminates the credit that had been taken in the original analysis for "special one of a kind checking" as a recovery factor, and the arbitrary factor of ten reduction for errors in the execution portion of the human action.

The staff has a remaining concern regarding the treatment of time in the revised post-initiator event analysis. The impact of time was modeled only indirectly in terms of opportunity for error recovery. Although the licensee did not take the significant credit for error recovery that had been taken in the original analysis, the CBDT method does not, in itself, analyze time-critical actions wherein the possibility for operator failure in decision making and performing an action within major time constraints (e.g., times in the range of seconds to 1 minute) is significant. However, other than ATWS events the revised analysis shows that there are few "short-term" actions with time on the order of 5, 10 and 20 minutes wherein actions are performed within the control room that are considered reasonable applications of the method.

Therefore, the staff does not believe that this possible shortcoming of the analysis would have prevented the licensee from identifying a vulnerability.

The licensee indicated that using the revised HRA values increased the total CDF approximately 24% over the frequency of 4.2E-5/ry identified for total core damage frequency in the original analysis. The revised HRA analysis resulted in increases in contributions from sequences involving large/medium LOCA (failure to switch over to recirculation), loss of service water (failure to diagnose, align, and start the essential service water) and transients with and without the power conversion system (failure to feed and bleed) and contributed to changes in the rankings of the contributions to the initiating events as noted above. The staff believes that the licensee, through the revised HRA, has gained a quantitative understanding of the contribution of human events to the CDF, and has improved its ability to discover vulnerabilities to severe accidents from human errors. Therefore, the staff finds the process used in the modified human reliability analysis consistent with the intent of Generic Letter 88-20. The staff encourages the licensee in future revisions of the Wolf Creek IPE to better document the process used to identify and select pre-initiator (miscalibration) events and to better treat time in the modeling of errors associated with actions that have to be performed within a very short time.

Based on the licensee's IPE process used to search for decay heat removal (DHR) and internal flooding vulnerabilities, and review of WCGS plant-specific features, the staff finds the licensee's DHR and flooding evaluation consistent with the intent of the USI A-45 (Decay Heat Removal Reliability) and USI A-17 (Systems Interactions in Nuclear Power Plants) resolutions respectively. No other specific unresolved safety issues (USIs) or generic safety issues (GSIs) were proposed for resolution as part of the WCGS IPE.

The licensee evaluated and quantified the results of the severe accident progression through the use of WCGS plant specific phenomenological evaluation papers and a small 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 WCGS conditional containment failure probabilities, the licensee estimated that early containment failure is 0.1%, late containment failure is 4% with overpressurization (due to steam generation or accumulation of non-condensible gases) or base-mat melt through being the primary contributors, and bypass is 0.2% with SGTR and interfacing systems LOCA sequences being the primary contributors. According to the licensee, the containment remains intact 95% of the time. The licensee's response to containment performance improvement program recommendations is consistent with the intent of GL 88-20 and the associated Supplement 3.

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

1. Ability to perform bleed and feed cooling.

2. 4 high pressure (2 charging and 2 safety injection) emergency core cooling system pumps to provide RCS injection and makeup flow.

3. Service water system flexibility and redundancy with dedicated standby essential service water system (ESW) pumps.

4. Ability to use ESW system as a source of water supply to the auxiliary feedwater pumps.

5. Eight hour battery capacity after shedding of selected DC loads.

6. Establishment of high pressure recirculation from the sump requires manual actions of the operators to align the discharge of the RHR pumps to the suction of the safety injection and/or the charging pumps.

In section 3.4.2 (Vulnerability Screening) of the IPE submittal, the licensee indicated that the results of the WCGS PRA were evaluated against the NUMARC Severe Accident Closure Guidelines (NUMARC 91-04). WCNOC stated that they have not identified any vulnerabilities at WCGS. However, the licensee identified several plant enhancements listed below that were being evaluated, and that, if implemented, would decrease the CDF. The licensee indicated that credit for only items 3 and 5 below, was taken in the IPE.

1. Installation of high temperature qualified RCP seal O-rings. The licensee indicated that, if the new O-rings were installed, that it would occur in early 1999, and estimated that they would reduce the CDF from 4.2E-5/ry to 3.7E-5/ry.

2. Replacement of the positive displacement charging pump with a third centrifugal charging pump. Actual installation of the pump will be performed after the eighth refueling outage during normal plant operation. The licensee indicates that if their assumption, that operation of this pump is not dependent on cooling water, is correct, the CDF may be reduced from 4.2E-5 to 3.6E-5/ry.

3. Provide a switch to bypass feedwater isolation in order to restore main feedwater. A modification is planned for the ninth refueling outage (fall 1997) which will provide this capability for all conditions. Full credit for this modification was mistakenly taken in the IPE based on a partial modification done in 1993. The licensee indicated that if, conservatively, no credit is taken for this modification, the CDF would increase about 19% from 4.2E-5 to 5.0E-5/ry.

4. Enhance emergency procedures to directly address total loss of component cooling water (CCW) and service water (SW) initiating events. Procedural changes have been made to Procedures OFN EG-004 (CCW System Malfunctions) and OFN EF-033 (Loss of Essential SW) to provide alternate cooling from other systems for lube oil cooling for the charging and safety injection pumps. The licensee estimates that if credit is taken for these enhancements, the CDF would be decreased by about 7% from 4.2E-5 to 3.9E-5/ry.

5. The licensee indicated that one enhancement related to the Station Blackout Rule that has been implemented and credited in the IPE was the shedding of selected DC loads to extend battery life up to eight hours. Without credit for load shedding the CDF would increase about 12% from 4.2E-5 to 4.9E-5/ry from increases in station blackout sequences.

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 GL 88-20 (and associated guidance NUREG-1335), and (2) the IPE results are reasonable given the WCGS 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 WCGS 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 the WCGS 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

WOLF CREEK GENERATING STATION INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT
(FRONT-END)