WOLF CREEK
NUCLEAR OPERATING CORPORATION

June 26, 2007

Terry J. Garrett
Vice President, Engineering

ET 07-0026

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Reference: 1) Letter ET 06-0038, dated September 27, 2006, from T. J. Garrett, WCNOC, to USNRC
2) Letter dated February 7, 2007, from C. Jacobs, USNRC, to T. J. Garrett, WCNOC

Subject: Docket No. 50-482: Response to NRC Requests for Follow-up Information Regarding Severe Accident Mitigation Alternatives for Wolf Creek Generating Station License Renewal Application

Gentlemen:

Reference 1 submitted Wolf Creek Operating Corporation's (WCNOC) application for renewal of the operating license for the Wolf Creek Generating Station (WCGS). The NRC staff identified in Reference 2, areas where additional information is needed to support the review of the Severe Accident Mitigation Alternatives.

A teleconference was conducted on March 20 to discuss the request for additional information in Reference 2. The NRC staff presented WCNOC with follow-up questions during the teleconference. WCNOC responses to the follow-up questions are provided in Attachment I.

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NRR

If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Kevin Moles at (620) 364-4126.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachment

cc: J. N. Donohew (NRC), w/a
V. G. Gaddy (NRC), w/a
B. S. Mallett (NRC), w/a
V. Rodriguez (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By____________________
Terry J. Garrett
Vice President Engineering

SUBSCRIBED and sworn to before me this 26th day of June, 2007.

NOTARY PUBLIC OFFICIAL SEAL STATE OF KANSAS
RHONDA L. TIEMEYER
MY COMMISSION EXPIRES
January 11, 2010

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2010

Questions and Responses from Teleconference Regarding the Analysis of Severe Accident Mitigation Alternatives (SAMAs) for the Wolf Creek Generating Station

Questions and Responses from Teleconference Regarding the Analysis of Severe Accident Mitigation Alternatives (SAMAs) for the Wolf Creek Generating Station

1. **The ER states that the initiating event contribution to core damage for the 1996 Updated IPE model was 6.3E-5. Yet in the letter ET 96–0034/96-0068 this value is shown as 6.19E-5. Clarify this discrepancy.**

RESPONSE:

The total Core Damage Frequency (CDF) value for the original Wolf Creek Generating Station (WCGS) probabilistic safety analysis (PSA) was 4.19E-05/year. The general content and results of this original WCGS PSA were submitted to the NRC as reflected in the WCGS Individual Plant Examination (IPE) (Letter WM 92-0152).

Two major model changes were made as a result of questions in the June 28, 1995 NRC Request for Additional Information (RAI), regarding the IPE Submittal. First, the Common Cause Failure events were revised to utilize generic common cause factor values. Second, the Human Reliability Analysis (HRA) values were revised to address a number of RAI comments specific to the HRA performed for the original WCGS PSA. The core damage frequency resulting from incorporation of these changes into the model was 6.19E-05/year. These changes are presented and described in Letter ET 96-0034. Letter ET 96-0068 included a description of the more significant changes that resulted in the 6.19E-05/year CDF value.

A final review of the new HRA values resulted in a revision to several of the HRA events. In addition, the final HRA resolved a number of instances where dependencies between HRA actions were not explicitly accounted for in the PSA model. These HRA action dependencies were described in the Letter ET 96-0034. Letter ET 96-0034 (Attachment page 2 of 69, 1st bullet) estimated that an explicit modeling of these HRA action dependencies would result in a CDF increase of approximately 4 percent over the 6.19E-05/year value. With the final HRA values and the inclusion of explicit HRA action dependencies, the resultant CDF value was 6.31E-05/year.

The 6.31E-05/year CDF value was not formally submitted to the NRC since it fell well within the 4 percent increase estimate of Letter ET 96-0034 and since none of the conclusions with regard to the significance of core damage sequences were changed. For WCGS utilization purposes, the WCGS PSA with the resultant CDF of 6.31E-05/year is considered the July 1996 model listed in the table in Appendix F, Section F.2.2 of the Environmental Report.

2. **In Section F.2.3.7, listed for the 1998 PSA model update, explain what is meant by the phrase, "Major active risk significant component groups."**

RESPONSE:

This question is under the Data Analysis section of the 1998 PSA Model Details. The following discussion gives an insight to the types of data changes that were made to Structures, Systems, and Components (SSC)s that are major active risk significant components.
The Data Analysis Notebook collected plant specific failure data for the following component groups:

- Major Safety Related Motor Operated Valves (MOV)s; Fail to transfer on demand
- Major Safety Related Pumps (Safety Injection (SI), Coolant Charging Pump (CCP), Auxilliary Feedwater (AFW), Residual Heat Removal (RHR), Containment Spray (CS), Component Cooling Water (CCW), Essential Service Water (ESW); Fail to start, Fail to run
- Diesel Generators; Fail to Start, Fail to run
- Turbine Driven AFW Pump; Fail to Start, Fail to run
- FCHV0312; Fail to Transfer on Demand
- ESW Traveling Screens; Fail to start, Fail to run

The motor driven pump (MDP) data was reviewed and the various pumps separately updated as appropriate. This addressed the concern that good data from the majority of pumps could potentially mask poorer failure data within one or two classes of pumps. Therefore, for the 1998 PSA update, pump failure data was considered separately by type of pump. The review resulted in the decision to separately Bayesian update the plant experience for the following pumps: AFW pump fail to run, charging pump fail to run and RHR pump fail to start, since the plant failure data indicated that they were failing at a higher rate than the other safety related pumps at Wolf Creek. Note that in the data analysis for "ALL MDPs" either failing to start or failing to run, the failures and start demands and run times for the MDAFW pumps, charging pumps and RHR pumps were included. While this gives greater weight to the failures that occurred for these three groups of pumps, it was considered that the types of failures which occurred could, in general, be experienced by any of the MDPs.

Infrequently tested components were also separately updated since they could reasonably be postulated to have a higher failure probability than would more frequently tested components. The only infrequently tested components updated were certain MOV. Therefore, for the purposes of the 1998 update, infrequently tested MOVs were updated as a separate component class. The failure data for the MOVs previously evaluated in the 1996 model was subdivided into data for frequently tested MOVs, and for infrequently tested MOVs (those MOVs which are tested on an 18 month frequency). One problem encountered in evaluating the MOV data was that the infrequently tested MOVs often receive multiple demands during shutdown conditions. Therefore the data encountered during shutdown (other than the initial stroke during the refueling) is more representative of frequently tested MOVs than it is of infrequently tested MOVs. This was addressed conservatively by considering only the first shutdown demand in determining the number of failures and demands on infrequently tested MOVs.

MOV stroke tests for PSA use would only be counted as a single demand (or alternatively that failures of the initial stroke be counted as two failures) to avoid biasing

the MOV failure data. Therefore surveillance tests were counted as a single demand in determining the number of valve demands used in the 1998 data update.

3. F&O TH-6 addresses issues associated with the definition of core damage. Describe the definition of core damage used in 2002 PSA Update.

RESPONSE:

The definition for core damage used in the WCGS Individual Plant Examination (IPE) reads: "core exit thermocouple temperature >1200°F for 30 minutes." This is essentially identical to a suggested definition in ASME RA-Sb–2005 Addenda to ASME RA-S–2002 Standard for PRA for NPP, SC-A2, "....code-predicted core exit temperature >1,200°F for 30 min using a code with simplified core modeling.".

The IPE core damage definition has been carried through the 2002 PSA Model Update.

4. In Section F.2.4, a table is provided that shows "Initiating Event Frequency Changes." Confirm that these numbers represent CDFs not initiating event frequencies.

RESPONSE:

Some of the values in the table were indeed core damage frequencies. The intended information in the subject table was Initiating Event Frequencies. The corrected Initiating Event Frequencies are shown in the response to Question 1.a of Letter ET 07-0006.

5. Provide the following information concerning the MACCS analyses:

a. Identify the units used in Table F.3-3 for Frequency, Conditional Dose within 50 Miles and Conditional Cost within 50 Miles.

b. Section F.3.5 states that meteorology data was collected from the WCGS meteorological monitoring program, but does not indicate where this data was collected. Clarify where the data is collected.

c. In Section F.3.3, it is stated that the core inventory used for the analysis was derived from the plant's safety analysis based on Westinghouse Letter SAP-99-145. Confirm that the resulting core inventory reflects the WCGS-specific fuel burnup/management as the plant is expected to be operated during the renewal period (including the power uprate). If this is not the case, evaluate the impact on population dose and on the SAMA screening and dispositioning if the SAMA analysis were based on the fission product inventory for the highest burnup and fuel enrichment expected at WCGS during the renewal period.

RESPONSE:

a. The Frequency is "per year," the Conditional Dose is "person-rem per year," and Conditional Cost is "dollars per year."

b. The data was collected from the on-site meterological tower.

c. Current core design practice remains consistent with the source term in Letter SAP-99-145. There are no plans to change the operating strategy at this time.

6. In the benefit analysis portion of SAMA 1, the ER states that the total CDF of SBO sequences SBOS02 through SBOS32 is 1.61E-06. This appears to be a typographical error. Verify that the CDF for these sequences should actually be 1.61E-05.

RESPONSE:

Yes, the sum of SAMA 1 SBO sequences SBOS02 through SBOS32 is 1.61E-05. The "E-06" value is a typographical error.