?

WOLF CREEK
NUCLEAR OPERATING CORPORATION

June 26, 2007

Terry J. Garrett
Vice President, Engineering

ET 07-0023

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
3) Letter ET 07-0006, dated April 20, 2007, from T. J. Garrett, WCNOC, to USNRC

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 (SAMA) analysis. WCNOC responded to the NRC request in Reference 3.

On May 23, 2007, the NRC provided by electronic mail, six follow-up questions regarding SAMA responses provided by WCNOC in Reference 3. A teleconference was conducted on May 24, 2007 to discuss the questions. All questions were resolved during the teleconference with the exception of numbers 1 and 3. WCGS responses to questions number 1 and 3 are provided in Attachment I.

A121

HRR

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 I WCNOC Response to NRC Requests for Follow-up Information Regarding Severe Accident Mitigation Alternatives

cc: J. N. Donohew (NRC), w/a
V. G. Gaddy (NRC), w/a
C. Jacobs (NRC), w/a
B. S. Mallett (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

Attachment I

Wolf Creek Nuclear Operating Corporation (WCNOC) Response to NRC Requests for Follow-up Information Regarding Severe Accident Mitigation Alternatives

WCNOC Follow-up Questions for Discussion - #1

In response to Question 1b (QU-9) it is stated that "internal flooding scenarios have not been included in PSA updates and that new internal flooding criteria may identify new human-induced floods through consideration of errors of commission." State how this could impact the SAMA analysis.

Response:

WCNOC's response to this question is based on the original response to question 1.b, QU-9, found in Reference 1. New criteria have since emerged from the 2005 Addenda of the ASME PRA Standard that may be factored in, in the future.

In spite of this new guidance, Wolf Creek Generating Station's (WCGS) compartmentalization is expected to minimize any new hypothetical flooding scenarios resulting from new evaluations. Based on our experience with the PRA model, flooding analysis and the extent of compartmentalization at WCGS, minimal impact on the Severe Accident Mitigation Analysis (SAMA) analysis is expected as a result of additional analysis to new requirements. The conclusions of the SAMA analysis would not be expected to change.

WCNOC Follow-up Questions for Discussion - #3

In response to Question 2b, it is stated that NUREG-1570 asserts that only 2 percent of the high pressure melt scenarios with dry steam generators would result in an induced SGTR and that the applicability of percentage is predicated on the conditions that the secondary side is not depressurized and that the RCPs are not operated. A review of NUREG-1570 found that for cases where a RCP Seal LOCA occurs that the TI-SGTR failure probability for 3 steam generators (4 SG scenario is not available) is 1.0. State how this increased failure likelihood on RCP seal failure was accounted for in the response to Question 2b.

NUREG-1570 includes failure probabilities for Transient Induced (TI)- Steam Generator Tube Rupture (SGTR) for a variety of situations. The conditions associated with the 2% failure are; 1) no Steam Generator (SG) depressurized and 2) no Seal loss of coolant accident (LOCA). As described in the original request for additional information (RAI) response, these are the most likely conditions that the plant would be in for this type of assessment. However, NUREG-1570 does show that for cases with depressurized Steam Generators combined with a cleared loop seal, the failure probability for TI-SGTR is 1.0. The inference from the NRC followup question is that a Seal LOCA will result in a cleared loop seal. It is important to note, however, that a failure probability of 1.0 also requires that the Steam Generators are depressurized.

A recent public meeting (ML071230212) was held jointly by the NRC and EPRI to discuss steam generator tube rupture. SCDAP/RELAP5 analyses were presented by the NRC to investigate various sensitivities associated with tube heatup. Included in the NRC presentation was a discussion of loop seal clearing and the impact on tube failure for a range of seal leakage rates. One of the conclusions from the NRC analysis was, that for Reator Coolant Pump (RCP) leakage at the cold leg centerline, no loop seal clearing was observed for leak rates of 300 gpm per pump and lower. In addition to clearing the loop seal, the lower downcomer skirt path must also clear to allow the hot gas to circulate throughout the primary system. Clearing this path required a seal leakage of 480 gpm per pump. For RCP shaft seal leakage located below the cold leg centerline, loop seals

were calculated to clear with leak rates greater than 120 gpm per pump. It is also important to note that the NRC analysis assumed 0.5 in^2 steam leakage from all steam generators resulting in the SGs being depressurized.

The original RAI response looked at the increase in the maximum averted cost risk assuming that 2% of all "Late Containment Failure" and "No Containment Failure" cases resulted in a SGTR. Another method to assess the impact of TI-SGTR is to review the major contributors to core damage frequency (CDF) from Table F.2.4 of the Environmental Report. Consistent with the discussion above, all cases with a Seal LOCA greater than 120 gpm per pump in combination with a depressurized steam generator will be assumed to result in a TI-SGTR (Failure probability = 1.0). Sequence identifier SBOS12 is defined as a Station Blackout with Auxilliary Feedwater (AFW) initially available, but assumed to be lost after a failure to recover AC power in 4 hours. This sequence type also assumes successful Reator Coolant System (RCS) cooldown and depressurization. For sequences of this type, it is not certain that the SG would remain in a depressurized state after loss of DC power. However, as a conservative assumption they will all be assumed to involve both an elevated primary side pressure in combination with a depressurized secondary side. This sequence group has a CDF contribution equal to 3.582E-6 per reactor year. Assuming that this entire frequency is then added to the existing SGTR frequency will result in an updated SGTR CDF of 3.75E-6. To conserve the total frequency, the SBOS12 CDF is subtracted from the "No Containment Failure" probability. Making this adjustment to the total SGTR probability will result in a SGTR dose-risk of 0.83 person-rem compared to the base value of 0.04. The economic cost risk for SGTR is increased from $72 to $1,625. While this represents a significant increase in the SGTR specific contribution, the following confirms no significant impact on the overall SAMA conclusions.

The maximum averted cost risk increases from the base value of $1,852,000 to $1,946,000. Based on a review of the non-cost beneficial SAMAs in the 95th percentile PRA case from the SAMA analysis, SAMA 4 is seen to be "not cost beneficial" by the smallest margin ($137,601). Even if all of the TI-SGTR risk could be mitigated by SAMA 4 in addition to its baseline mitigated risk (which is not physically possible for SAMA 4), it would still not be cost beneficial by $43,601 ($137,601-$94,000=$43,601). Given that the margins are even larger on the remaining "not cost beneficial" SAMAs, it can be concluded that changes to the treatment of TI-SGTRs would not have an impact on the conclusions of the WCGS SAMA analysis.

References:

1) Letter ET 07-0006, dated April 20, 2007, from T. J. Garrett, WCNOC, to USNRC.