WOLF CREEK
NUCLEAR OPERATING CORPORATION

Terry J. Garrett
Vice President Engineering

January 18, 2011

ET 11-0001

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

Reference:
1) Letter ET 10-0014, dated April 13, 2010, from T. J. Garrett, WCNOC, to USNRC
2) Letter dated August 18, 2010, from B. K. Singal, USNRC, to M. W. Sunseri, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Regarding License Amendment Request to Revise Technical Specification Table 3.3.2, "Engineered Safety Feature Actuation System Instrumentation" (TAC NO. ME3762)"
3) Letter ET 10-0028, dated October 13, 2010, from T. J. Garrett, WCNOC, to USNRC
4) Letter dated November 24, 2010 from B. K. Singal, USNRC, to M. W. Sunseri, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Regarding License Amendment Request to Revise Technical Specification Table 3.3.2, "Engineered Safety Feature Actuation System Instrumentation" (TAC NO. ME3762)"
5) Letter ET 10-0038, dated December 21, 2010, from T. J. Garrett, WCNOC, to USNRC

Subject: Docket No. 50-482: Additional Information Related to Second Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"

Gentlemen:

Reference 1 provided Wolf Creek Nuclear Operating Corporation's (WCNOC) application to revise Technical Specification (TS) Table 3.3.2-1, Function 8.a., of TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." Reference 2 provided a request

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for additional information related to the application. Reference 3 provided WCNOC's response to the request for additional information. In Reference 4, the Nuclear Regulatory Commission (NRC) staff indicated that the staff has reviewed Reference 3 and determined that additional information is needed to complete the review. Reference 5 provided WCNOC's response to the second request for additional information (Reference 4).

In a January 3, 2011, teleconference call with the NRC Project Manager, it was identified that the response to Question 2 of Reference 5 did not address the events in Updated Safety Analysis Report (USAR) Section 15.6.2, "Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary that Penetrate Containment," and 15.6.3, "Steam Generator Tube Rupture (SGTR)." The Attachment provides additional information for the response to Question 2 of Reference 5.

The additional information does not expand the scope of the application as originally noticed, and does not impact the conclusions of the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the Federal Register (75 FR 33844).

In accordance with 10 CFR 50.91, a copy of this submittal is being provided to the designated Kansas State official.

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4084, or Mr. Gautam Sen at (620) 364-4175.

Sincerely,

Terry J. Garrett

TJG/rlt

Attachment: Additional Information Related to Response to Second Request for Additional Information

cc: E. E. Collins (NRC), w/a
T. A. Conley (KDHE), w/a
G. B. Miller (NRC), w/a
B. K. Singal (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 18th day of January, 2011.

Gayle Shepheard
Notary Public

GAYLE SHEPHEARD
Notary Public - State of Kansas
My Appt. Expires 7/24/2011

Expiration Date 7/24/2011

Additional Information Related to Response to Second Request for Additional Information

Reference 1 provided Wolf Creek Nuclear Operating Corporation's (WCNOC) application to revise Technical Specification (TS) Table 3.3.2-1, Function 8.a., of TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation." Reference 2 provided a request for additional information related to the application. Reference 3 provided WCNOC's response to the request for additional information. In Reference 4, the Nuclear Regulatory Commission (NRC) staff indicated that the staff has reviewed Reference 3 and determined that additional information is needed to complete the review. Reference 5 provided WCNOC's response to the second request for additional information (Reference 4).

In a January 3, 2011, teleconference call with the NRC Project Manager, it was identified that in the response to Question 2 of Reference 5, the response did not address the events in Updated Safety Analysis Report (USAR) Section 15.6.2, "Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary that Penetrate Containment," and 15.6.3, "Steam Generator Tube Rupture (SGTR)." Provided below is additional information related to the response to Question 2 of Reference 5. The specific NRC question is provided in italics.

2. *To address the acceptability of the proposed TS deletion of the turbine trip and feedwater isolation in Mode 3, the licensee stated in its response to RAIs 2 and 3 that the turbine trip function is not required to obtain acceptable results for Chapter 15 analyses. Also, the response to RAI 4 stated that neither the turbine trip nor the feedwater isolation functions are required to obtain acceptable results within the non-loss-of-coolant accident (LOCA) Chapter 15 analyses. Please provide the bases to support the above statements in the responses to RAI 2 and 3 and RAI 4 for each of the events in FSAR Chapter 15, and show that none of the "events analyzed in Modes 1 and 2 would become more severe if the events were analyzed in Mode 3 (or below) assuming the proposed P-4 function are defeated."*

Response:

<u>Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary that Penetrate Containment (USAR Section 15.6.2)</u>

The most severe pipe rupture with regard to radioactivity release during normal plant operation was determined to be a rupture of the Chemical and Volume Control System (CVCS) letdown line at a point outside of the containment. For such a break, the reactor coolant letdown flow would have passed sequentially from the cold leg and through the regenerative heat exchanger and letdown orifices. The letdown orifice reduces the letdown line pressure from 2,235 psig to less than 600 psig outside containment during normal plant operation when letdown flow is maintained at 120 gpm. Increase in flow will occur due to a rupture of the letdown line downstream of the orifices. It has been determined that the occurrence of a complete severance of the letdown line would result in a loss of reactor coolant at the rate of 141 gpm. Based on this maximum leakage rate, the dose consequence calculation has confirmed that the radiological consequences of a postulated letdown line rupture outside containment do not exceed a small fraction of the exposure guidelines as set forth in 10 CFR Part 100.

The event is limiting in MODE 1 since the flow rate out of the break will be maximized at full power operation. The analysis presented in the USAR only addresses the short-term phase of the event to the point of termination of primary coolant loss from the letdown line rupture. The event is terminated by operator action to close an isolation valve in the affected path within 30 minutes. It is noted that the system transient response, including reactor trip, or any of the P-4 functions are not modeled in the analysis. Thus, the proposed deletion of the P-4 functions in MODE 3 does not impact the analysis of the event and the analysis presented in USAR Section 15.6.2 remains bounding.

Steam Generator Tube Rupture (SGTR) (USAR Section 15.6.3)

The consequences of SGTR depend largely upon the ability of the operator to take the necessary actions to terminate the primary to secondary leakage. If the leakage continues for an extended period of time, the secondary side of the steam generator may become filled and water may enter the steamline. As a result, the release of liquid through the secondary side safety/relief valves to the atmosphere may occur that could result in an increase in the radiological doses.

An analysis was performed to determine the thermal and hydraulic transient for the limiting SGTR scenario, which results in steam generator overfill and water release through the steam generator safety valve, with a consequential failure of the safety valve following water release. The offsite radiation doses were calculated based upon the mass releases from the transient analysis and the site specific meteorological parameters. Since operator actions are required to mitigate the consequences of an SGTR, the offsite radiation doses are necessarily dependent upon the time required for the operator to complete the recovery actions. The major SGTR recovery actions include identification and isolation of the ruptured steam generator, cooldown and depressurization of the Reactor Coolant System (RCS), and termination of safety injection. These actions are designed to equalize the RCS and the ruptured steam generator pressures, and thus to terminate the primary to secondary leakage.

Following the occurrence of the SGTR, the primary to secondary leakage causes the pressurizer level and the RCS pressure to decrease. As the RCS pressure continues to decrease, automatic reactor trip occurs on a low pressurizer pressure, and is followed shortly thereafter by a turbine trip. Because of the assumed loss of offsite power, the Steam Dump System will not be available, and the secondary side pressure increases rapidly after reactor trip until the steam generator atmospheric relief valves and/or safety valves lift to dissipate the energy. After reactor trip, the RCS pressure continues to decrease and the safety injection is automatically initiated on low pressurizer pressure signal. Due to the assumed loss of offsite power at the reactor trip, normal feedwater flow is terminated and auxiliary feedwater flow is initiated.

Although a turbine trip is modeled to occur on reactor trip, the turbine trip is not used to mitigate the event. Delaying or deleting the actuation of turbine trip would be a benefit to the analysis since the turbine would provide an additional heat removal path. The additional heat energy removed through the turbine would enhance the RCS depressurization and consequentially reduce the primary to secondary leakage. The primary means for mitigation of this event are the ability of operators to take the necessary action to equalize the primary and secondary pressures to terminate the primary to secondary leakage. Thus, the P-4 interlock function to trip turbine on reactor trip does not provide any mitigating effects for this event. Therefore, the proposed deletion of the P-4 functions in MODE 3 does not impact the event and the current analysis presented in USAR Section 15.6.3 remains bounding.

References:

1. WCNOC Letter ET 10-0014, "Application to Revise Technical Specification 3.3.2, "Engineered Safety Feature Actuation System Instrumentation," Table 3.3.2-1," April 13, 2010.

2. Letter from B. K. Singal, USNRC, to M. W. Sunseri, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Regarding License Amendment Request to Revise Technical Specification Table 3.3.2, "Engineered Safety Feature Actuation System Instrumentation" (TAC NO. ME3762)," August 18, 2010.

3. WCNOC Letter ET 10-0028, "Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"," October 13, 2010.

4. Letter from B. K. Singal, USNRC, to M. W. Sunseri, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Regarding License Amendment Request to Revise Technical Specification Table 3.3.2, "Engineered Safety Feature Actuation System Instrumentation" (TAC NO. ME3762)," November 24, 2010

5. WCNOC Letter ET 10-0038, "Response to Second Request for Additional Information Regarding License Amendment Request to Revise Technical Specification 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation"," December 21, 2010.