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

November 24, 2010

Mr. Matthew W. Sunseri
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: 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)

Dear Mr. Sunseri:

By letter dated April 13, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101100391), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted a license amendment request to revise Table 3.3.2-1 of Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System Instrumentation." Specifically, the amendment would add footnote (m) to Function 8.a. of TS Table 3.3.2-1 to identify the enabled functions and the applicable modes for the Reactor Trip, P-4 interlock function. By letter dated August 18, 2010 (ADAMS Accession No. ML102180064), the U.S. Nuclear Regulatory Commission (NRC) staff requested additional information, and the licensee responded by letter dated October 13, 2010 (ADAMS Accession No. ML102920142).

The NRC staff has reviewed the WCNOC response and determined that additional information identified in the enclosure to this letter is needed in order for the staff to complete its review. A draft copy of the enclosed request for additional information was provided to Ms. Diane Hooper of your staff via e-mail on November 11, 2010. Mr. Steve Wideman of WCNOC informed the NRC staff on November 19, 2010, that a conference call to discuss the request for additional information is not needed. On November 23, 2010, Mr. Wideman agreed to provide the response by December 23, 2010.

If you have any questions, please contact me at 301-415-3016 or balwant.singal@nrc.gov.

Sincerely,

for

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure
As stated

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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO REVISE TS 3.3.2, "ENGINEERED SAFETY FEATURE

ACTUATION SYSTEM INSTRUMENTATION" TABLE 3.3.2-1

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

By letter dated April 13, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101100391), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted a license amendment request to revise Table 3.3.2-1 of Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System Instrumentation." Specifically, the amendment would add footnote (m) to Function 8.a. of TS Table 3.3.2-1 to identify the enabled functions and the applicable modes for the Reactor Trip, P-4 interlock function. According to the WCNOC letter, this LAR is being submitted to prevent unnecessary cycling of the main feedwater isolation valves, and to preclude confusion regarding applicable modes and plant conditions for compliance with TS Table 3.3.2-1, Function 8.a. By letter dated August 18, 2010 (ADAMS Accession No. ML102180064), the U.S. Nuclear Regulatory Commission (NRC) staff requested additional information and the licensee responded by letter dated October 13, 2010 (ADAMS Accession No. ML102920142).

The NRC staff has reviewed the WCNOC response to the request for additional information (RAI) and determined that the following additional information is needed to complete its review:

1. In its letter dated October 13, 2010, the licensee stated in response to RAI 2 that "the potential RCS [reactor coolant system] cooldown caused by the turbine trip is bounded by the cooldown caused by the MODE 2 HZP SLB [Hot Zero Power Steamline Break] event." In the Final Safety Analysis Report (FSAR) Chapter 15, there are five cooldown events:

    a. Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature (Updated FSAR Section 15.1.1)

    b. Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (Section 15.1.2)

    c. Excessive Increase in Secondary Steam Flow (Section 15.1.3)

    d. Inadvertent Opening of a Steam Generator Atmospheric Relief or Safety Valve (Section 15.2.4)

    e. Steam System Pipe Failure (Section 15.1.5)

Enclosure

Please discuss the effects of the turbine trip and main feedwater isolation function on each of the above cooldown events and show that with the proposed TS deletion of the turbine trip and feedwater isolation functions of the P-4 interlock, the consequences from the above events initiating from Mode 3 through 6 are bounded by the FSAR Chapter 15 analysis for corresponding events initiating from Mode 1 or 2.

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."

3. The licensee's response to RAI 4 discussed an analysis of the steam line break (SLB) event initiating from Mode 3 below P-11 with no feedwater isolation. Please provide a description of the SLB analysis. The requested information should include the results of the analysis, a sequence of the events, a discussion of the methods and computer code used in the analysis, and a discussion of the compliance with the restrictions and limitations in the NRC safety evaluation report approving the methods and code. Please identify the key parameters considered and discuss the bases of the selection of the values (including measurement uncertainties and parameter fluctuation around the normal values) for the initial plant conditions that minimize the margin to acceptable limits.

4. In Section 2.4 of its letter dated October 13, 2010, the licensee stated that the reactor trip P-4 interlock turbine trip function is also not credited for Appendix K small-break LOCA (SBLOCA) analyses. This could result in an additional increase in RCS depressurization during the SBLOCA, which is non-conservative. Besides, the LOCA blowdown load forces are much less for the limiting small breaks, so not tripping the turbine for SBLOCA appears to be inappropriate. Secondly, the licensee's statement in Section 3.4 of its letter dated October 13, 2010, that "SBLOCA events are typically insensitive to small changes in secondary side heat removal" is not always true. For plants with core uncover and high peak centerline temperatures (PCTs) for small breaks in the range 0.05 to 0.1 ft^2, small changes in secondary heat removal can have an appreciable impact on PCT because lower RCS pressures of 25 to 50 pounds per square inch absolute over several hundred seconds during the initial portion of a limiting SBLOCA can reduce PCT substantially, as a result of the additional high-pressure safety injection flow injected during this period due to this lower RCS pressure. Please show the impact of failure of turbine trip on the limiting SBLOCA and compare the results with the case where the turbine is tripped.

5. In its response to RAI 1, the licensee indicated that "the subject circuitry does not provide a required safety function" and that the feedwater isolation signal bypass is not expected to be implemented more than once per year; therefore, required bypassed and inoperable status indication per NRC Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," is not warranted.

   Please confirm that inclusive of all the purposes identified for bypassing the feedwater isolation function during times when the feedwater system is expected to be operating (e.g., when performing procedures GEN 00-006, "Hot Standby to Cold Shutdown," STS AE-201, "Feedwater Chemical Injection Inservice Valve Test," GEN 00-002, "Cold Shutdown to Hot Standby, and any others, if needed) that jumpers for bypassing the feedwater isolation would not be installed more than once per year.

If you have any questions, please contact me at 301-415-3016 or balwant.singal@nrc.gov.

Sincerely,

/RA by N. Kalyanam for/

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure
As stated

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ADAMS Accession No.: ML103260174 * Memo dated 11/15/2010 **via email

OFFICE	NRR/LPL4/PM	NRR/LPL4/PM	NRR/LPL4/LA	NRR/DSS/SNPB/BC
NAME	LGibson	BSingal (JRHall for)	JBurkhardt	AMendiola**
DATE	11/22/10	11/23/10	11/22/10	11/3/10
OFFICE	NRR/DE/EICB/BC	NRR/DSS/SRSB	NRR/LPL4/BC	NRR/LPL4/PM
NAME	WKemper*	AUlses**	MMarkley	BSingal (NKalyanam for)
DATE	11/15/10	11/12/10	11/24/10	11/24/10

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