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

May 24, 2011

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 THE FIRE PROTECTION PROGRAM (TAC NO. ME4757)

Dear Mr. Sunseri:

By letter dated September 22, 2010, as supplemented by letter dated November 22, 2010, the Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC) for the Wolf Creek Generating Station (WCGS). The proposed amendment consists of changes to the approved fire protection program as described in the WCGS Updated Safety Analysis Report (USAR). Specifically, the licensee is requesting approval for a deviation from a commitment to certain technical requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix R, Section III.L.1, "Alternative and dedicated shutdown capability," as described in Appendix 9.5E of the WCGS USAR. WCNOC has proposed to revise USAR Table 9.5E-1 to include information on reactor coolant system process variables not maintained within those predicted for a loss of normal AC [alternating current] power.

The NRC staff has reviewed the licensee's letters dated September 22 and November 22, 2010, and has determined that further information is needed to complete its review of the LAR. In order to complete our review in a timely manner, the NRC staff requests that WCNOC provide its response to the enclosed request for additional information by June 30, 2011.

If you have any questions regarding this action, please contact me at (301) 415-4032.

Sincerely,

James R. Hall

James R. Hall, 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 CHANGE THE FIRE PROTECTION PROGRAM

COMMITMENTS IN THE UPDATED SAFETY ANALYSIS REPORT

WOLF CREEK GENERATING STATION

WOLF CREEK NUCLEAR OPERATING CORPORATION

DOCKET NO. 50-482

By letter dated September 22, 2010, as supplemented by letter dated November 22, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML102720417 and ML103340290, respectively), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) requested approval from the U.S. Nuclear Regulatory Commission (NRC) to amend the operating license for the Wolf Creek Generating Station (WCGS). The proposed amendment consists of changes to the approved fire protection program as described in the WCGS Updated Safety Analysis Report (USAR). Specifically, the licensee is requesting approval for a deviation from a commitment to certain technical requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix R, Section III.L.1, "Alternative and dedicated shutdown capability," as described in Appendix 9.5E of the WCGS USAR. WCNOC has proposed to revise USAR Table 9.5E-1 to include information on reactor coolant system (RCS) process variables not maintained within those predicted for a loss of normal AC [alternating current] power.

The NRC staff has determined that the following information is needed to complete its review.

1. Page 6 of Attachment 1 to the licensee's letter dated September 22, 2010, indicated that each of the 16 control room fire scenarios included in evaluation SA-0-006, Revision 1, was modeled using RETRAN-3D.

   Please confirm that the RETRAN-3D code used was previously approved by the NRC and provide a reference to the NRC safety evaluation report (SER) approving the code. Also, please explain how this application complies with each of the restrictions and conditions specified in that SER. Please identify and provide justification for any changes from the NRC-approved version of RETRAN-3D.

2. Attachment II to the licensee's letter dated November 22, 2010, presents the results of analyses for four control room fire scenarios, designated as Scenarios 1, 1A, 2, and 2A.

   Please identify the major input initial conditions and the worst single failure assumed in the analysis, discuss the bases used to select the numerical parameters, and demonstrate that the numerical values used are conservative, with consideration of the uncertainties and fluctuations around the nominal values. Please confirm that the values applied result in the minimum departure from nuclear boiling ratio, maximum core exit temperature (CET), and greater void formation in the RCS.

Enclosure

Please discuss the sequence of events for the representative case of the above 4 scenarios with the power-operated relief valve (PORV) opening duration of 3 minutes. Also, please describe, for Scenarios 1A and 2A, the system response (related to temperature, pressure, core flow rate and void fraction) in the time sequence up to 3,600 seconds, and show that there is no unexplainable physical phenomenon predicted.

Please provide a table of the sequence of events for the representative case including setpoints for the systems credited in the analysis, and address the acceptability of the associated setpoints. Please provide justification for the non-safety-related systems, if any, used in the analysis for mitigating the consequences. In addition, please discuss the thermo-hydraulic conditions for the fluid (steam or water) released from the PORV during the events analyzed and justify that the PORV can be closed on demand.

3. Figures 2-5 and 2A-5 in Attachment II to the licensee's letter dated November 22, 2010, present the RCS temperature responses for Scenarios 2 and 2A, respectively. The figures show that during the period from 2,800 to 3,500 seconds, the predicted core inlet temperature, reactor vessel average temperature, and CET fluctuate around the stable values. Please explain the phenomenon of the RCS temperature fluctuation.

   The figures also show that from 3,500 to 3,600 seconds, the predicted RCS temperatures continue to increase. Please expand the analysis beyond 3,600 seconds to demonstrate that the RCS temperatures will decrease and stabilize over an extended period of time, assuring the long-term cooling capability.

   For Scenarios 2 and 2A, the analysis assumes that no loss of offsite power will occur. Please explain the assumption or automatic actuation credited in the analysis that results in a reactor coolant pump coast-down as shown in Figures 2-2 and 2A-2 for the decreasing core mass flow rate in terms of the system elapsed time.

4. As shown in Figures 1-7, 1-8, 1A-7, and 1A-8 of the licensee's letter dated November 22, 2010, the analysis predicts that the steam bubble formation will occur in the upper core and upper reactor vessel head regions for Scenarios 1 and 1A with the PORV opening duration assumed to be equal to or greater than 3 minutes.

   Please describe and address the acceptability of the model used for predicting bubble formation. Please describe how the results of the analysis show that there are no steam bubbles carried into the RCS hot legs, steam generator (SG) U-tubes, RCS cold legs, down comer and lower core regions, and that any steam bubbles that do accumulate at the top of the SG U-tubes do not block the natural recirculation flow for decay heat removal.

5. Page 15 of Attachment 1 to the licensee's letter dated September 22, 2010, stated, in part, that

   > The effect of some core voiding later in the transient does not present a challenge to fuel integrity as long as natural circulation is maintained and core exit temperatures remain below 712°F. ... For these scenarios,

continuous positive core mass flow rates demonstrate that natural circulation was maintained and core exit temperatures less than 600°F confirm that cladding integrity was not challenged.

Please provide the following information:

a. The technical bases including applicable experimental data to support the adequacy of use of the CETs of 712 °F as an acceptable criterion for the fuel cladding integrity;

b. A discussion of the calculated CET of less than 600 °F to show that it bounds the CETs in the peripheral and central regions in the reactor core for all analyzed scenarios; and

c. A discussion of the natural circulation (NC) model used in the analyses to show the acceptance of the use of the NC model for Scenarios 1 and 1A with voiding occurred in the upper core and upper reactor head regions (shown in Figures 1-7, 1-8, 1A-7, and 1A-8, of the licensee's letter dated November 22, 2010).

6. Please explain how the operators will close the PORV within the assumed 3-minute time period, considering that it is designed for steam and may experience liquid flow in some scenarios. Please identify and justify the use of any non-safety-related systems relied upon to close the PORV.

May 24, 2011

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Sincerely,
/RA/

James R. Hall, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

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