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

February 25, 2013

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 – WITHDRAWAL OF LICENSE AMENDMENT REQUEST RE: DEVIATION FROM FIRE PROTECTION REQUIREMENTS (TAC NO. MF0427)

Dear Mr. Sunseri:

By letter dated December 20, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13002A146), Wolf Creek Nuclear Operating Corporation submitted a license amendment request for Wolf Creek Generating Station (WCGS) that proposed changes to the approved fire protection program as described in the WCGS Updated Safety Analysis Report (USAR).

By letter dated February 5, 2013 (ADAMS Accession No. ML13050A039), you requested to withdraw the application from U.S. Nuclear Regulatory Commission (NRC) review. The NRC acknowledges your request. NRC staff activities on the review have ceased and the associated Technical Assignment Control (TAC) number has been closed. The application was not noticed in the *Federal Register*.

The NRC staff notes that its review to date has identified that your application did not provide technical information in sufficient detail to enable the staff to complete its detailed review. The deficiencies in your submittal were summarized in a telephone conference between me and members of the NRC staff with M. Westman and other members of your staff on January 29, 2013. Specific deficiencies in your submittal identified by the NRC staff are listed below. You may consider requesting a pre-application meeting with the staff before making any future re-submittal. The staff considers that Item 1 below is the major challenge to be addressed in your submittal. The other items were identified by the staff during a more detailed review of the submittal:

1. In general, the submittal lacks sufficient discussion of the plant's defense-in-depth for fire protection, including a discussion of the accident scenario prior to dependence on the thermal-hydraulic analysis; the fire detection and suppression measures for the impacted areas; the information regarding postulated fire scenarios, ignition sources, and location of important circuits affected by the fire; and a discussion of the timeline of a potential scenario that would result in entry into procedure OFN-RP-017. In addition, the assumptions concerning unaffected equipment lack sufficient detail (e.g., see submittal Appendix V, which shows a diagram of the control room and states that, since the cabinets are physically separated, the remaining trains of the solid state protection system (SSPS) would be unaffected. This only addresses the cabinets and does not

address whether the cables involved in SSPS would be affected.). More specifically, the submittal lacks:

a. A description of the postulated fire scenarios, ignition sources, fuel loading, and target cables (e.g., such as those that would open power-operated relief valves (PORVs), inhibit block valves from closing, controlling centrifugal charging pumps, cause loss of offsite power). The NRC staff noted that a limited discussion of the bounding fire testing per NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets: Part 1: Cabinet Effects Tests," April 1987 (ADAMS Accession No. ML060590316), was included in the licensee's submittal, but it was not specific to the plant configuration.

b. A discussion of alternatives if defense-in-depth does not prevent the transient. The NRC staff recommends that the licensee refer to recent NRC exemptions (e.g., see the March 2010 exemption for James A. FitzPatrick Nuclear Power Plant available at ADAMS Accession No. ML100340670) which provide a defense-in-depth analysis. In the licensee's submittal, the alternative appears to be reliance on manual actions to prevent unrecoverable conditions, even if they are outside of 10 CFR Part 50, Appendix R assumptions.

2. The revised Assumption 3-A-4 removes the discussion of the loss of offsite power and automatic starting of the emergency diesel generators. It is unclear why the discussion of loss of offsite power is being removed from the assumption.

3. No loss of the automatic function of the feedwater isolation signal is assumed, based on cabinet separation.

a. A diagram and discussion (submittal page 11 of 34) focuses on cabinet separation, but does not provide cable routing information or a justification of why cable routing is not important. The reference to Regulatory Guide 1.75, Revision 2, "Physical Independence of Electric Systems," September 1978 (ADAMS Accession No. ML003740265), is not sufficient, based on the available fire damage information relating to cables.

b. It appears that the only failure mode is assumed to be damage due to heat, and there is no discussion of smoke damage in adjacent cabinets (e.g., see the example of smoke damage documented in an NRC letter dated March 12, 2012, to Omaha Public Power District (ADAMS Accession No. ML12072A128)).

4. Accident Scenario 1 (submittal page 17 of 34) describes that a PORV is "stuck open" and is manually closed at 180 seconds.

a. It is unclear whether "stuck open" is the correct terminology or whether "spuriously open" is intended.

b. Table 7.1 (submittal page 58 of 102) describes that the PORV is closed by isolating control power. However, if the spurious actuation of the PORV is due to a hot short, then the electrical current may be provided by a source other than the designated control power. In that case, it is unclear how procedure step C2 assures PORV closure.

c. The required time to complete procedure step C2 is 180 seconds. Scenario time T=0 is tripping the reactor, subsequently followed by control room evacuation, plant-wide announcements, implementation of the Emergency Plan, traveling to the emergency locker outside the auxiliary shutdown panel, and the remainder of the procedure. Procedure step C2 appears to occur following RP-017, step 6.b. If the reactor trip occurs concurrently with the spurious PORV opening, it is unclear that all of the actions leading up to procedure step C2 (OFN-RP-017, steps 1 through 6, and the initial step in Attachment C) can be performed in 180 seconds.

d. In submittal pages 57-59 of 102, procedure step C2 isolates power to close the PORV, but step C3 requires the operator to obtain a copy of the procedure. It is unclear how step C2 is performed without a procedure.

5. Scenario 3A includes the assumption that the centrifugal charging pumps (CCPs) are available, but no fire analysis or separation (either cable or cabinet) analysis is provided to assure that the CCPs are available for the same fire scenario where the steam generator (SG) "A" atmospheric relief valve is stuck open.

6. Use of the RETRAN computer code

a. The submittal lacks a discussion of the adequacy of the Chexal-Lelloche drift flux model used in RETRAN for calculating mass distributions on the steam generator secondary side and simulating vapor collection in the upper regions of the reactor coolant system (RCS) for conditions with boiling occurrence.

b. The submittal lacks a discussion of the adequacy of the use of RETRAN to show that the natural recirculation (based on single- or two-phase flow) can be maintained for conditions when boiling occurs.

7. Acceptance criteria of the thermal hydraulic analysis

The licensee used the criterion that the average RCS hot-leg temperature of less than 630 degrees Fahrenheit (°F) to show no core damage to occur.

a. The submittal lacks a discussion of the bases of the criterion used to show no core damage discussed above, and explain why the departure-from-nucleate-boiling ratios and fuel rod centerline temperatures are not calculated to ensure the integrity of the fuel and cladding by showing satisfaction of the respective acceptable limits.

b. The submittal lacks a discussion for cases 1, 1A, 1C, and 3A of Attachment 1 to the submittal the reactor coolant pump seal leakage model used in the analysis for conditions with the RCS pressure equal to or greater than 2250 pounds per square inch absolute and the cold-leg temperature equal to or greater than

550 °F. The submittal lacks justification if no reactor coolant pump seal leakage model is considered in the analysis.

8. Sequence of Events. The following information was lacking in the submittal:

   a. (1) A table listing the sequence of events for cases 1, 1A, 1C, and 3A, with specifications of the setpoints for those events that relied on automatic actuation; (2) A discussion of how instrumentation uncertainties are considered and the operator action times for those events that relied on operator actions; and (3) For the operator actions, a description to show why the actions can be achieved within the operator action times.

   b. (1) A list of the assumptions and values of the plant initial conditions used in the analyses, and justification that those assumptions and initial conditions are representative of WCGS; and (2) A discussion of the uncertainties for the initial values of the plant parameters used in the analyses, or a discussion showing why the uncertainties are not considered.

As a potential reference for any future re-submittal, the NRC staff received a similar request by STP Nuclear Operating Company (STPNOC) and provided an evaluation (see ADAMS Accession Nos. ML12222A023 and ML12297A331, respectively). The staff understands that STPNOC intends to resubmit its application later this year. If you have any questions, please contact me at 301-415-2296 or via e-mail at fred.lyon@nrc.gov.

Sincerely,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

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/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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ADAMS Accession No.: ML13039A064 *Previously concurred

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