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

December 4, 2014

Mr. Adam C. Heflin
President, Chief Executive Officer,
and Chief Nuclear Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KS 66839

SUBJECT: WOLF CREEK GENERATING STATION – REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE AMENDMENT REQUEST TO REVISE THE FIRE PROTECTION PROGRAM RELATED TO ALTERNATIVE SHUTDOWN CAPABILITY (TAC NO. MF3112)

Dear Mr. Heflin:

By letter dated November 21, 2013, Wolf Creek Nuclear Operating Corporation (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) a license amendment request to revise the fire protection program as described in the Updated Safety Analysis Report (USAR) for the Wolf Creek Generating Station. The licensee's proposed changes would incorporate a revised alternate shutdown methodology into the USAR.

The NRC staff has reviewed the information provided in your application and determined that additional information is required in order to complete its formal review. The enclosed questions were provided to Mr. S. Wideman of your staff on December 4, 2014. Please provide a response to the enclosed questions within 45 days of the date of this letter.

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-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO REVISE THE FIRE PROTECTION PROGRAM

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

By letter dated November 21, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13331A728), Wolf Creek Nuclear Operating Corporation (the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) a license amendment request to revise the fire protection program as described in the Updated Safety Analysis Report (USAR) for the Wolf Creek Generating Station. The licensee's proposed changes would incorporate a revised alternate shutdown methodology into the USAR.

The NRC staff requests the following additional information in order to complete its review of the request.

Reactor Systems Branch Questions:

SRXB-RAI-1

In Section 3.7.5 of the submittal, it states that maximum operator response times were, when possible, set as less than or equal to 80 percent of the time-sensitive action required time. This section also claims that the 80 percent threshold partially accounts for instrumentation uncertainties. The effects of time-sensitive human actions are important to determining the acceptability of the presented sequences. Please provide:

a. A list of time-sensitive actions in Evaluation SA-08-006 which were identified by procedure AL 21-017 to require between 80 percent and 100 percent of the four time-sensitive action required time to complete.

b. A list of conservative assumptions purposefully included in evaluation SA-08-006 to account for uncertainties in plant response.

In addition,

c. Please explain if the 80 percent threshold also is meant to account for delays due to operator errors. If yes, explain how the 80 percent threshold was developed. The NRC staff normally accounts for operator error by postulating credible errors for specific tasks and adding the time margin required to recover from the worst case credible error. If a default value is to be used instead of a task-specific margin, the time margin should be 100 percent (i.e., the estimated time required should be doubled; see Appendix B of NUREG-1852, "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire," October 2007 (ADAMS Accession No. ML073020676), for details). If the margin

of 100 percent is used, the NRC staff would expect your threshold for further action to be set at 50 percent, not 80 percent.

SRXB-RAI-2

Please provide an explanation of the intended purpose of Figures 3 and 5 of the submittal; as presented, the differences between the plots of the individual sequences are indistinguishable.

SRXB-RAI-3

The discussion of the applicability of the Chexal-Lellouche correlation is difficult to follow in its current form. Please provide a clear explanation of the following:

a. The parameter ranges in Table 3 of Attachment 1 extend outside the Chexal-Lellouche experimental database. Please explain what the plant conditions are in both the primary and secondary systems during which the Chexal-Lellouche correlation is applied to the RETRAN-3D calculation.

b. The licensee stated that the Chexal-Lellouche correlation could be used to determine when and if binding occurs in the steam generator tubes, but the evaluations in SA-08-006 do not predict significant steam accumulation in the steam generator tubes. Please clarify when and where the Chexal-Lellouche correlation is or is not used in each of the 24 scenarios.

c. The analysis of the submittal's sequences 1, 1A, and 1C show that voids develop in the core regions and then move to the steam generators, where the voids collapse in the first steam generator tube volume. The lowest void fraction in the steam generator tubes is reported as 0.0 in Table 3 of Attachment 1. Please explain whether the void fraction reported in Table 3 is a rounded value, or whether the voids collapse in the steam generators so quickly that no voids exist at the bottom of the steam generator.

d. Please explain the purpose of applying the Chexal-Lellouche to the primary system calculations, if no voids exist in the bottom of the steam generator tubes.

SRXB-RAI-4

In Section 3.5 of the application, the licensee states, in part, that, "there is reasonable assurance that the fire will remain in the cabinet of origin and will not spread." This conclusion has important implications as to the acceptability of the selected sequences in the supporting analysis. If the fire were to spread beyond the cabinet of origin, the impact upon systems could be significantly different than that of a single cabinet fire with significant implications for the thermal-hydraulic reasons to the accident scenarios. Thus, validation of the submittal conclusion that a fire would not spread beyond the cabinet of origin is important. Please explain or illustrate whether or not the electrical cabinet separation requirements for adjacent cabinets, that resulted from the tests documented in NUREG/CR-4527, "An Experimental Investigation of Internally Ignited Fires in Nuclear Power Plant Control Cabinets," April 1987 (ADAMS Accession No. ML060960351), are consistent with the actual main control room cabinets at the site (e.g.,

the separation between SB038 and SB037 cannot be compared to the requirements established in NUREG/CR-4527).

SRXB-RAI-5

Multiple spurious auctions are important in the progression of accident sequences. Please provide the following:

a. A discussion regarding how single or multiple spurious actuations have been considered in each of the accident scenarios, consistent with the requirements in Regulatory Guide 1.189, Revision 2, "Fire Protection for Nuclear Power Plants," October 2009 (ADAMS Accession No. ML092580550), Sections 5.4.1 and 5.4.4, which specify that such actuations should be considered after control has been transferred from the control room to the alternative or dedicated shutdown system and after control of the plant has been achieved.

b. A discussion regarding the consideration of spurious actuations that could defeat the alternate safe shutdown system (e.g., spurious actuations that could negate the successful isolation of the main feedwater system or the chemical injection system). This information will be used by the NRC staff to validate the input assumption used in the supporting analysis.

SRXB-RAI-6

Please explain how fuel thermal conductivity degradation is accounted for in the analysis.

SRXB-RAI-7

Please justify the core axial power shape that is used in the analysis.

SRXB-RAI-8

Please explain if loss of feedwater without offsite power may cause the pressurizer to overfill.

SRXB-RAI-9

Please list the operator actions that will be taken in the control room prior to evacuation due to a fire, and identify how these operator actions are incorporated into the analysis used in support of this request.

Fire Protection Branch Questions:

The amendment request included changes from the licensee's commitments to certain technical requirements of Title 10 of the *Code of Federal Regulations* Part 50, Appendix R, Section III.L, as documented in the WCGS Fire Hazards Analysis Report. The commitments are related to crediting the automatic feedwater isolation system for closure of the main feedwater isolation valves (MFIVs) and/or the main feedwater regulating valves (MFRVs) and the MFRV bypass valves to terminate main feedwater flow and prevent steam generator overfill in the event that a

fire necessitates the evacuation of the control room.

FP-RAI-01 Operator Manual Actions

Table 7.1 within E-1F9915, "Design Control Document For OFN RP-017, Control Room Evacuation," provides a detailed evaluation for each step in OFN RP-017. For the required actions from outside the control room, OFN RP-017 does not specify whether the operator manual actions have been evaluated for feasibility and reliability, e.g., per NUREG-1852 "Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire (NUREG-1852)," October 2007 (ADAMS Accession No. ML073020676).

Provide a discussion of basis for the feasibility and reliability of operator manual actions performed outside of the control room.

FP-RAI-02 Multiple Spurious Actuation

Drawing E-1F9915, Section 2.2, "Assumptions," item 2.2.3 states:

> Prior to transfer of control to the Auxiliary Shutdown System only a single spurious actuation is assumed to occur at a time, except in the case of two redundant valves in a high/low pressure interface line. All potential spurious actuations are mitigated/prevented using OFN RP-017 but timing is based on the spurious actuations occurring one at a time, or two at a time in the case of high/low pressure interface lines.

Regulatory Guide 1.189 "Fire Protection for Nuclear Power Plants," Revision 2, Section 5.4.4 states in part:

> After control of the plant is achieved by the alternative or dedicated shutdown system, single or multiple spurious actuations that could occur in the fire-affected area should be considered, in accordance with the plant's approved FPP.

Please justify the reasoning for assuming that after control of the plant is achieved from the alternative location that the timing of spurious actuations should be based on the spurious actuations occurring one at a time for non-high/low pressure interface lines.

December 4, 2014

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If you have any questions, please contact me at 301-415-2296 or via e-mail at fred.lyon@nrc.gov.

Sincerely,

/RA/
Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure
Request for Additional Information

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