WOLF CREEK
NUCLEAR OPERATING CORPORATION

Jaime H. McCoy
Vice President Engineering

November 14, 2017
ET 17-0025

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

References: 1) Letter ET 17-0001, dated January 17, 2017, from J. H. McCoy, WCNOC, to USNRC

2) Electronic mail dated October 17, 2017, from B. K. Singal, USNRC, to W. T. Muilenburg, WCNOC, "Request for Additional Information - License Amendment Request (LAR) for Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term Wolf Creek Generating Station (CAC No. MF 9307)"

Subject: Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analyses Including Adoption of Alternative Source Term

To Whom It May Concern:

Reference 1 provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). The proposed amendment would support transition to the Westinghouse Core Design and Safety Analysis methodologies. In addition, the amendment request included revising the WCGS licensing basis by adopting the Alternative Source Term radiological analysis methodology in accordance with 10 CFR 50.67, "Accident Source Term." Reference 2 provided a request for additional information related to the application. The Attachment provides WCNOC's response to the request for additional information.

The additional information does not expand the scope of the application and does not impact the no significant hazards consideration determination presented in Reference 1.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," a copy of this submittal is being provided to the designated Kansas State official.

A001
NRR

This letter contains no commitments. If you have any questions concerning this matter, please contact me at (620) 364-4156, or Cynthia R. Hafenstine at (620) 364-4204.

Sincerely,

Jaime H. McCoy

JHM/rlt

Attachment: Response to Request for Additional Information

cc: K. M. Kennedy (NRC), w/a
B. K. Singal (NRC), w/a
K. S. Steves (KDHE), w/a
N. H. Taylor (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Jaime H. McCoy, 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 Jaime H McCoy
Jaime H. McCoy
Vice President Engineering

SUBSCRIBED and sworn to before me this 14th day of November, 2017.

STATE OF KANSAS NOTARY PUBLIC
GAYLE SHEPHEARD
My Appointment Expires
July 24, 2019

Gayle Shepheard
Notary Public

Expiration Date 7/24/2019

Response to Request for Additional Information

Reference 1 provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). The proposed change replaces the WCNOC methodology for performing core design, non-loss-of-coolant-accident (non-LOCA) and LOCA safety analyses to the standard Westinghouse methodologies for performing these analyses, and associated TS changes. Reference 1 would also revise WCGS TS's and the Updated Safety Analysis Report Chapter 15 radiological consequence analyses using an updated accident source term consistent with Title 10 of the Code of Federal Regulations (10 CFR), Section 50.67, "Accident source term." Reference 2 provided a Nuclear Regulatory Commission (NRC) request for additional information related to the application. The specific NRC question is provided in italics.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Chapter 7, "Instrumentation and Controls," August 2016 (ADAMS Accession No. ML16020A049), defines the acceptance criteria for this review. Standard Review Plan Chapter 7 addresses the requirements for instrumentation and control systems in light-water nuclear power plants. The regulatory requirements and guidance which the NRC staff considered in its review are as follows:

- *10 CFR 50.36(c)(1)(ii)(A) requires in part that where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor.*
- *10 CFR 50.36 (c)(2)(i) requires that the TSs include limiting conditions for operation (LCOs) for equipment required to ensure safe operation of the facility. When an LCO for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.*
- *10 CFR 50.36 (c)(3) states TS Surveillance Requirements (SRs) relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. 10 CFR 50.36, "Technical specifications," states, "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section." Specifically, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether a limiting condition for operation is required to be included in the TS.*
- *10 CFR 50.55a(h) requires that the protection systems must meet the requirements in Institute of Electrical and Electronics Engineers (IEEE) Std. 279–1968, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," or the requirements in IEEE Std. 279–1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," or the requirements in IEEE Std. 603–1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995.*

EICB-RAI 3

a. *In the LAR, Enclosure IV, WCNOC requested to add Surveillance Requirement (SR) 3.3.7.6 to TS 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation*

Instrumentation," Function 2 (Automatic Actuation Logic and Actuation Relays and Function 3 (Control Room Radiation - Control Room Air Intakes) of Table 3.3.7-1. Enclosure IV of the LAR identifies the new SR, and provides the basis for adding it. In particular, WCNOC is requesting to add this SR to ensure that the time delay used in the AST model is bounded. Enclosure IV states that the acceptance criteria for the CREVS actuation instrumentation is ≤ 60 seconds.

EICB-RAI 3-1

Please provide additional information, such as calculations, to support the proposed CREVS actuation instrumentation response criteria.

Response:
The 60 second response time is derived from the design response times from the components necessary to complete the CREVS actuation, plus additional margin. The time accounts for the detector (response time is on the order of microseconds), process racks (response time is approximately 0.3 seconds), Solid State Protection System (SSPS) (response time is approximately 2 seconds), Emergency Diesel Generator (EDG) start time (response time is approximately 12.2 seconds), and damper stroke time (longest damper stroke time is 30 seconds). The combined response time (approximately 45 seconds), is then rounded up to 60 seconds to ensure that the analysis time bounds the actual combined component response time.

Regarding the dose calculations, the component response time is accounted for by including a 60 second delay from when the control room air intake setpoint is reached to when the CREVS lineup is credited. Therefore the time considered in the analyses bound the actual combined component response time.

The proposed SR requires verification of the "Control Room Ventilation Isolation Engineered Safeguard Features (ESF) RESPONSE TIMES are within limits." The TS does not include the acceptance criteria for the CREVS actuation system.

EICB-RAI 3-2

Please explain how operators will determine if the acceptance criteria to determine that the response time is within limits?

Response: In order to verify that the CREVS response time credited in the dose analyses bounds the response time of the actual plant components, the response time of each component (except the detector which is explicitly discussed in the response to EICB-RAI 3-3 within this letter) will be measured at the frequency specified by Surveillance Requirement (SR) 3.3.7.6. Once the response time for each of the applicable plant components has been measured, the overall response time for a safety train will be calculated by summing each applicable individual measured time. The resulting total response time will then be compared to the analysis value of 60 seconds to ensure that the actual plant response is bounded by the time modeled in the analyses.

With regards to explicitly not listing the response time within the WCGS TS, this convention is consistent with other response time testing SRs. Specifically, the WCGS TS SRs document that response time testing will be performed, but they do not explicitly list the limiting time within

TS. However, the corresponding response time is listed in the TS Bases. For example, this format is use for the following TS SR:

SR 3.3.1.16 (Acceptance Criterion listed in Table B 3.3.1-2 of TS Bases)
SR 3.3.2.10 (Acceptance Criterion listed in Table B 3.3.2-2 of TS Bases)
SR 3.3.5.4 (Acceptance Criterion listed in Section B 3.3.5 of TS Bases)
SR 3.3.6.6 (Acceptance Criterion listed in Section B 3.3.6 of TS Bases)

Consistent with this convention, the overall CREVS allowable response time of 60 seconds will be documented within the TS Bases (Markups provided in Section 11 of Enclosure IV of Reference 1).

The proposed SR also includes a note to exclude the radiation monitor detectors from ESF response time testing. This document states that this note is necessary "because of the difficulty associated with generating an appropriate radiation monitor detector input signal."

EICB-RAI 3-3

Please explain why there is "difficulty associated with generating an appropriate radiation monitor detector input signal." Also, please justify that the exclusion of radiation monitoring response time will provide an accurate measure of ESF response times.

Response: The Control Room Air Intake radiation monitor detectors have response times on the order of microseconds. In order to test the detector response time, a test would need to first create a condition where the detector could not observe a source at time zero and then introduce an observable source to the detector and accurately measure the time for the detector to output a corresponding signal. Due to the fact that the detector will respond to the source on the order of microseconds, it would not be feasible to test the installed detector response time to an accuracy greater than that of the actual response time. Furthermore, since the overall acceptance criterion is 60 seconds, the actual detector response time (on the order of microseconds) would be lost in the rounding when calculating the overall value.

It is worth noting that this convention is consistent with SR 3.3.1.16, which contains the following note: "Neutron detectors are excluded from response time testing." As documented in Section B 3.3.1 of the TS Bases, "This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response." Additionally, the Wolf Creek implementation of SR 3.3.1.16 is consistent with Standard TS (Reference 3). Specifically, the note for SR 3.3.1.16 in Reference 3 states, "Neutron detectors are excluded from response time testing."

Furthermore, the exclusion of the detectors from response time testing is consistent with the implementation of SR 3.3.7.6 by the Callaway Nuclear Plant, Unit 1 (Safety Evaluation documented in Reference 4). As documented on page 12 of the Safety Evaluation documented in Reference 4:

"For SR 3.3.7.6, the licensee also proposed (1) a note that radiation monitor detectors are excluded from response time testing, and (2) a frequency of 18 months on a staggered test basis for performing the surveillance. The note is based on the fact that

radiation monitor detectors are not response time tested because of the difficulty in generating an appropriate radiation detector input signal and the principles of radiation detector operation ensure a virtually instantaneous response. The frequency is the same as for other instrumentation response time SRs in the TSs, such as SRs 3.3.1.16, 3.3.2.10, 3.3.5.4, and 3.3.6.6, which all have the 18 months on a staggered test basis. Based on this, the staff concludes that the proposed note and frequency for SR 3.3.7.6 are acceptable."

Finally, the practice of excluding radiation detectors from response time testing is discussed in GL 93-08 (Reference 5). Specifically, the following discussion is provided within Reference 5:

"The surveillance requirements specify that the response time of each trip function is to be demonstrated to be within its limit at the specified frequency and do not reference the tables of response time limits. Therefore, the surveillance requirements specified in this manner need not be modified to implement this change. However, a footnote in the table of response time limits for the RTS states that neutron detectors are exempt from response time testing. To retain this exception, which is stated in the table being removed from the TS, the surveillance requirements for the RTS should be modified to add the following statement:

Neutron detectors are exempt from response time testing."

Therefore, it is acceptable to exclude the radiation monitoring response times due to the fact that the principles of radiation detector operation ensure a virtually instantaneous response, and excluding the detectors is consistent with SR 3.3.1.16 in the WCGS TS, SR 3.3.1.16 of the Standard TS, the implementation of SR 3.3.7.6 by the Callaway Nuclear Plant, Unit 1, and GL 93-08.

b. *In the LAR, Enclosure VI, WCNOC explained that they will implement a design modification to supply the CREVS control room isolation dampers with battery power. WCNOC noted that this modification is necessary to support implementation of AST.*

EICB-RAI 4

Please describe the logic to isolate the CREVS. This description should include a discussion of how and when battery backed power will insure CREVS isolation.

Response: There are two CREVS control room isolation dampers per safety train. Each safety train has an isolation damper on the recirculation duct (coming from the control room) and on the supply duct (going to the control room). Makeup to the control room is from the control building. A simplified diagram of the control room HVAC system is provided in Figure 1.

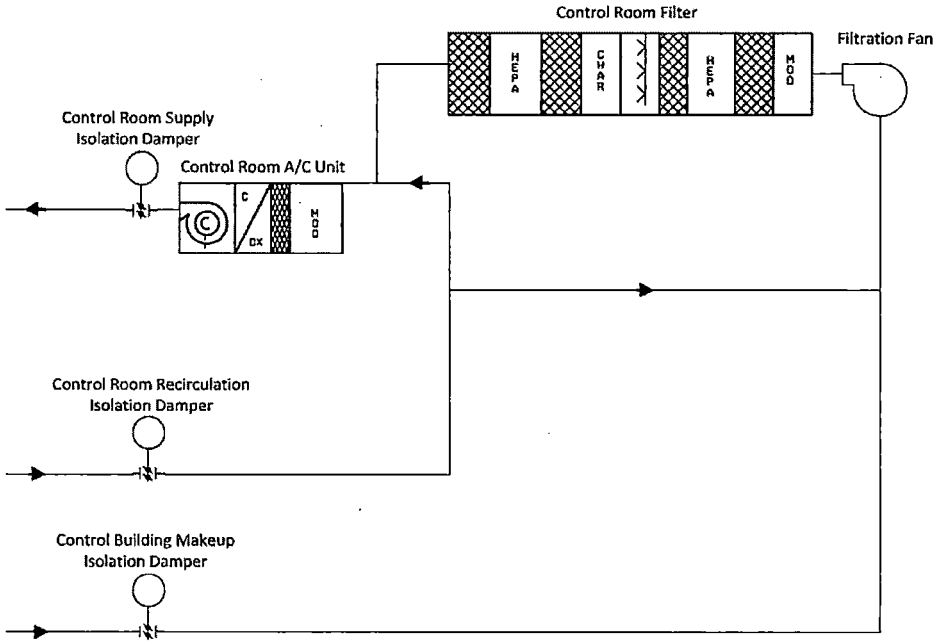

Figure 1: Simplified Control Room HVAC Diagram

The control room isolation dampers (labeled as Control Room Supply Isolation Damper and Control Room Recirculation Isolation Damper in Figure 1) are interlocked with the Control Room A/C Unit. Thus, if the Control Room A/C Unit is running, the two dampers are open. Likewise, if the Control Room A/C is off, the dampers are closed. The dampers are interlocked only with the Control Room A/C unit and there is no additional control for the dampers.

Currently, the dampers are powered from the same safety related source as the associated Control Room A/C unit. This source receives power from offsite sources if available or an EDG for a Loss of Offsite Power (LOOP) scenario. Depending on the single failure, it was possible for the isolation dampers to lose power and fail in the open position. The proceduralized response to this condition was to dispatch a local operator to locally isolate the affected train of CREVS.

In order to eliminate the requirement to dispatch a local operator, the design modification will change the control room isolation damper's power supply to safety-related battery power during all modes of operation. In order to support the modification, the battery load calculations have been assessed to ensure adequate battery load is available to support the planned modification. The control circuitry will not be changed. Thus, if a single failure, such as the failure of a filtration fan, were to occur in a CREVS train following an accident, the control room operators will have the capability to isolate the affected CREVS train from the control room by turning off the corresponding Control Room A/C Unit. Likewise, for the case where power is lost to the Control Room A/C Unit, such as a loss of the associated EDG, the isolation dampers will automatically close since they will receive a signal to close (interlocked with the Control Room A/C Unit) and they will be supplied with safety-related battery power.

Isolation damper position is indicated in the control room, and is powered from the same power source as the dampers. Thus, with or without this change, the position of the isolation dampers can be monitored from the control room as long as the dampers have power.

When applicable, such as following a reactor trip and a potential radioactive release, the control room operators will utilize emergency operating procedures to verify that CREVS has been actuated and is in the appropriate lineup. If a portion of the CREVS train has failed, the operators will take appropriate actions to isolate the train if needed. For the limiting single failure (failure of a filtration fan), the dose analyses model a 90 minute delay from the time when CREVS actuates to when it is isolated from the control room. This time delay is consistent with the current licensing basis time critical action (discussed in Section 15A.3 of the Updated Safety Analysis Report); however, following the implementation of the design modification, the action will be easier to accomplish as it can be completed remotely from the control room rather than having to dispatch a local operator.

Likewise, if the failure results in the loss of power to the Control Room A/C Unit, such as a failure of the associated EDG, the control room isolation dampers will close automatically (i.e., without operator action) and the control room operators will only need to verify that the system has responded appropriately. Thus, the modification will reduce operator burden and eliminate the potential of having to dispatch a local operator to locally isolate a failed CREVS train.

References:

1. Letter ET 17-0001, dated January 17, 2017, from J. H. McCoy, WCNOC, to USNRC

2. Electronic mail from B. K. Singal, USNRC, to W. T. Muilenburg, WCNOC, "Request for Additional Information – License Amendment Request (LAR) for Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term Wolf Creek Generating Station (CAC No. MF9307)," October 17, 2017. ADAMS Accession No. ML17291A710.

3. NUREG-1431, Revision 4, Volume 1, "Standard Technical Specifications - Westinghouse Plants," April 2012.

4. "Callaway Plant, Unit 1 – Issuance of Amendment Re: Equipment Hatch and Emergency Air Lock Open During Core Alterations or Movement of Irradiated Fuel Assemblies Inside Containment (TAC No. MB3605)," September 9, 2002. ADAMS Accession No. ML022620599.

5. Generic Letter (GL) 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits," December 29, 1993.