WOLF CREEK
NUCLEAR OPERATING CORPORATION

Jaime H. McCoy
Vice President Engineering

July 13, 2017
ET 17-0016

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

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

2) Electronic mail dated June 14, 2017, from B. K. Singal, USNRC, to W. T. Muilenburg, WCNOC, "Request for Additional Information - License Amendment Request 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

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.

A121
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

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
Vice President Engineering

SUBSCRIBED and sworn to before me this 13th day of July, 2017.

GAYLE SHEPHEARD
My Appointment Expires
July 24, 2019

_____
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 request for additional information related to the application. The specific NRC question is provided in italics.

Instrument and Controls Branch (EICB)

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 TS 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 TS 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.*

1. EICB-RAI 1

The LAR would replace the Allowable Value (AV) for Table 3.3.1-1 Function 10, "Reactor Coolant Flow – Low," for "Normalized Flow." This replacement is necessary for consistency with the assumptions of the new safety analysis methodology (i.e., the use of Normalized Flow, instead of design loop flow). WCAP-18083-P, Revision 0, "Westinghouse Revised Thermal Design Procedure Uncertainty Calculations for the Wolf Creek Generating Station," November 2016 (Enclosure 1 to letter dated March 22, 2017) defines the "Normalized Flow" as the reactor coolant system (RCS) flow normalization to the RCS flow calorimetric. However, it is not clear how the formula describes in the enclosure are used.

Please describe how procedurally the normalization process is carried out. The 12 hour surveillance test pertains to RCS Flow is SR 3.4.1.3, "Verify RCS total flow rate is > 3.71 x 104 gpm [gallons per minute] and greater than or equal to the limit specified in the COLR [core operating limit report]." Alternately, please provide the NRC staff with a copy of the 12 hour surveillance procedure.

Response: Two separate normalization processes are discussed within EICB-RAI 1. First, additional information is requested for the normalization process used to establish the Normalized Flow listed in TS Table 3.3.1-1 Function 10, "Reactor Coolant Flow – Low." Second, additional information is requested for how the normalization process is procedurally carried out for the 12 hour surveillance test.

<u>Reactor Coolant Flow – Low normalization process:</u>

Regarding the normalization process used to establish the Normalized Flow listed in TS Table 3.3.1-1 Function 10, "Reactor Coolant Flow – Low," the process is as follows. First, the output data for each of the Reactor Coolant System (RCS) flow transmitters is collected at full flow conditions, which is equivalent to 100% of indicated loop flow. This data is then used to calibrate the RCS flow transmitters in accordance with TS Surveillance Requirement (SR) 3.3.1.10. This process maintains the transmitter calibrated span to 120% of nominal flow and normalizes the output of each transmitter to full flow conditions.

After the transmitter calibration is addressed by calibrating the RCS flow transmitters to their corresponding full flow output voltage, the process cards are calibrated in accordance with SR 3.3.1.10 in order to verify that the Reactor Coolant Flow – Low bistable for each RCS flow loop will actuate once a voltage equivalent to 89.9% of the indicated loop flow is reached. Note: The nominal TS setpoint is 89.9% of indicated loop flow conditions and the Allowable Value is 88.9% of indicated loop flow conditions; therefore, the desired calibration value is set to the nominal value rather than the allowable value. This completes the normalization process for the Reactor Coolant Flow – Low trip function.

<u>SR 3.4.1.3 Normalization process:</u>

With regards to the normalization process of the 12 hour surveillance test SR 3.4.1.3, "Verify RCS total flow rate is > 3.71 x 10^4 gpm [gallons per minute] and greater than or equal to the limit specified in the COLR [Core Operating Limits Report]," the process is as follows. First, as part of the flow calorimetric process the RCS flow is measured. The flow calorimetric process is outlined in Enclosure II of the License Amendment Request, WCAP-18083-P. During this measurement, the reading of each of the 12 RCS flow control board indicators is recorded and

then averaged. This averaged indicator RCS flow reading (%) corresponds to the RCS measured flow (gpm) determined by the calorimetric.

Next, the calculated average reading for the control board indicators (%) is normalized to the minimum measured flow value.

Note EICB-RAI 1 lists a value of 3.71×10^4 gpm; however, while this is the historical value, the minimum measured RCS flow DNB Limit is being increased to 3.76×10^4 gpm as shown in Attachment V of the License Amendment Request.

This normalization is accomplished by calculating the ratio of the minimum measured flow (gpm) to the RCS measured flow (gpm). The resulting ratio is then multiplied by the calculated average of the 12 RCS flow control board indicators (%) to obtain the average control board indicator reading (%) corresponding to the minimum measured flow. This normalization process is performed on an 18 month cycle specific frequency to meet SR 3.4.1.4.

For example:

Assume the RCS measure flow is determined via the flow calorimetric to be 391,700 gpm.

Assume the average of the 12 control board indicators is 100.25% flow.

The ratio of minimum measured flow to RCS flow is 376,000/391,700 = 0.960

The control board indicator reading corresponding to minimum measured flow will be equal to

100.25% (0.960) = 96.2%

Once the average control board indicator reading corresponding to minimum measured flow is calculated, the control room shift log procedure is revised with the updated value. This completes the normalization process for SR 3.4.1.4.

The control room shift log procedure is then used to measure the control board indicator readings, average them, and compare them to the control board indicator reading corresponding to minimum measured flow to verify the SR 3.4.1.3 requirement is satisfied (i.e., in the example above, the average reading is verified to be above 96.2%). The control room shift log procedure requires that the RCS flow be verified once every 8 hours, to ensure that the 12 hour surveillance frequency is met.

Note that since the minimum measured flow value will not be changed until after the License Amendment Request (LAR) has been approved, the procedures implementing the normalization processes have not yet been revised with the new minimum measured flow value. Once the LAR is approved, the procedures will be issued with the updated value as part of the implementation effort.

2. EICB-RAI -2

The proposed amendment would add a new TS LCO 3.1.9, "RCS Boron Limitations <500°F." This modification necessities modification of the requirements for RTS trip

Function 2.b, Power Range Neutron Flux - Low, in TS Table 3.3.1-1. Specifically, the Applicability for reactor trip system (RTS) trip Function 2.b will be revised and new Conditions V, W, and X will be added to LCO 3.3.1. The LAR states that Condition V will be similar to existing Condition E. However, the NRC staff notes the end state for the plant Condition V will require the initiation of actions aimed at precluding an uncontrolled rod cluster control assembly bank withdrawal event from occurring or providing sufficient shutdown margin should this event occur while the existing condition E does not contain this requirement.

Please provide additional information to justify inclusion of this additional requirement.

Response: Condition V and Condition E are similar in that the required action is to either place the one inoperable channel in trip or require that the plant be placed in a condition where the trip function is not needed to protect the core from events applicable to the inoperable channel.

As discussed in the background section of TS Limiting Condition for Operation (LCO) 3.1.9 of the proposed TS Bases (Attachment IV to the LAR), TS LCO 3.1.9 addresses the concern identified in Nuclear Safety Advisory Letter (NSAL)-00-016 (Reference 3). Specifically, Reference 3 discussed that the primary reactor trip function (the Power Range Neutron Flux - Low trip) assumed in the analysis of an Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Low Power or Subcritical Condition event (RWFS) may not be operable at Reactor Coolant System (RCS) temperatures significantly below the hot zero power T-avg due to calibration issues associated with shielding caused by the cold water in the downcomer region of the reactor vessel. Thus, the additions of TS LCO 3.1.9 and the associated changes to TS Table 3.3.1-1 are in response to Reference 3.

Regarding Condition V, the purpose of the associated requirements is to put the plant in a condition where the trip function is not needed to protect the core from the RWFS event by either of the options provided; namely:

V.2.2.1 Initiate action to fully insert all rods

<u>AND</u>

V.2.2.2 Initiate action to place the Rod Control System in a condition incapable of rod withdrawal

<u>OR</u>

V.2.3 Initiate Action to borate the RCS to greater than all rods out (ARO) critical boron concentration

Since the RWFS event would be prevented in either of these conditions, the plant would no longer be in a condition where the Power Range Neutron Flux – Low Trip would be needed. Thus, Condition V and Condition E are similar in that both require an inoperable channel be

placed in trip, or require that the plant be placed in a condition where the trip function is not needed to protect the core from the RWFS event.

The addition of TS LCO 3.1.9 and the associated TS Table 3.3.1-1 Condition V requirements requested in this LAR are consistent with the addition of TS LCO 3.1.9 previously requested by Reference 4 and subsequently approved for the Callaway Nuclear Plant. The Safety Evaluation (Reference 5) stated the following in regards to Condition V:

> *The Required Action E.2 in the existing Condition E is to put the plant in a condition where the trip function is not required to perform its safety function. For the new Condition V, the licensee has proposed to change the second required action in the existing Condition E to either the required action 2.a or 2.b above, either of which puts the plant in a condition where the trip function is not needed to protect the core from the RWFS event. This is to say that either (1) having all control rods in and the rod control and placing the rod control system in a condition where it is not capable of rod withdrawal or (2) borating the RCS to greater than the ARO critical boron concentration will protect the core from the RWFS event. Based on this, the NRC staff concludes that the proposed Condition V for an inoperable [Power Range Neutron Flux]-low trip function is sufficient to protect the reactor core in Mode 1 (below the P-10 interlock) and Mode 2 (with keff > 1.0) and is, therefore, acceptable. The proposed completion time is the same as that for Required Action E.2; therefore, the licensee is not changing this requirement.*

In conclusion, the additional requirements of Condition V will either preclude an uncontrolled RCCA bank withdrawal event from occurring or provide sufficient shutdown margin if such an event were to occur. Therefore, the additional requirements of Condition V will ensure that the plant is in a condition where the trip function is not required to perform its safety function.

It is worth noting that the addition of TS LCO 3.1.9, and associated TS Table 3.3.1-1 changes, is in accordance with [Technical Specification Task Force] TSTF-453-T, Revision 1, as stated in Attachment I of the LAR. Callaway's submittal and associated TS changes were used as the precedent for the Wolf Creek TS changes, as Callaway's submittal ultimately resolved the issue identified in Reference 3.

Hydrology and Meteorology Branch (RHM1)

3. RHM1-RAI-1

> *10 CFR 50.67(b)(2)(i) requires a licensee seeking to revise its current accident source term in design basis radiological consequence analyses to provide an evaluation of the consequences of applicable design basis accidents to demonstrate that adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions. Methods acceptable to the NRC staff for determining atmospheric dispersion factors (or X/Q values) in support of design-basis control room radiological habitability assessments at nuclear power plants can be found in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003 (ADAMS Accession No. ML031530505).*
>
> *RG 1.194 provides guidance on the use of the ARCON96 atmospheric dispersion model for determining X/Q values to be used in design basis evaluations of control room*

radiological habitability analyses. Section 3.2 of RG 1.194 states that in order to determine source-receptor bounding combinations, it is necessary to consider the distance, direction, release mode, and height of the various release points to the environment in relation to the various control room intakes. Additional parameters, such as those used in establishing plume rise, may need to be considered in determining the bounding combination.

Section 4.1.2, "Atmospheric Dispersion Factors – Control Room and Technical Support Center, "Full Scope Implementation of Alternative Source Term," (Enclosure IV to letter dated January 17, 2017) describes the development of the X/Q values used in the control room (CR) and technical support center (TSC) radiological analysis. Tables 4.1.2-1(a), 4.1.2-1(b), and 4.1.2-2 in Enclosure 4 provide the ARCON96 input parameters for the Emergency Control Room Air Intake, Normal Control Room Air Intake, and the TSC Air Intake, respectively. The staff has identified 4 release heights in these tables that appear to conflict with each other. Please resolve the following potential conflicts by either updating the Enclosure 4 tables or justifying the use of the release heights listed below.

Source Release Heights

		Source			
Table No.	Receptor	Unit Vent Stack	MSSVs/ARVs[1] Vent	Turbine-Driven AFW[2] Exhaust Vent	Radwaste Building
4.1.2-1(a)	Emergency CR Air Intake	66.25 meters (m)	34.29 m	13.87 m	6.10 m
4.1.2-1(b)	Normal CR Air Intake	66.17 m	34.29 m	14.05 m	17.07 m
4.1.2-2	TSC Air Intake	66.25 m	35.71 m	13.87 m	2.84 m

Response: The differences in the release height from 4 sources identified by the NRC are in general a result of an assumption used in the determination of distance and height input to ARCON96 in order to minimize the resultant distance used by ARCON96 in the calculation of X/Q. The assumption is stated in Assumption 3 in section 4.1.2.3 "Assumptions and Acceptance Criteria" of the LAR. The assumption says,

> *"The shortest source-to-receptor distances are determined from the closest point on the perimeter of the source to the closest point on the perimeter of the receptor for all cases. Slightly different release heights from the same source may be used for different receptors to achieve the shortest distances."*

The four source locations listed with inconsistent release heights are: (1) the Unit Vent Stack, (2) the Main Steam Safety Valve/Atmospheric Relief Valve (MSSV/ARV) Vents, (3) the Turbine-Driven Auxiliary Feedwater (AFW) Exhaust Vent, and (4) the Radwaste Building. These sources are large in size and, for some, their geometry can be complicated. They are treated as point sources in the ARCON96 calculation. The height of these point sources can vary from case to case depending on the relative direction between the source and the receptor in order to minimize the distance between the source and the receptor.

The discrepancies for each source location are addressed individually.

Unit Vent Stack

For releases from the Unit Vent Stack to the Emergency Control Room (CR) Air Intake and Technical Support Center (TSC) Air Intake, the release elevation that was used was the lowest point on the Unit Vent opening screen. This release location appropriately minimizes the slant distance between the source and the receptor. For the release from the Unit Vent Stack to the Normal CR Air Intake, the release elevation was decreased by 4 inches to conservatively account for uncertainty in the reference drawing. This decreased elevation yields additional conservatism in the slant distance between the source and the receptor location.

MSSV and ARV vents

The MSSV and ARV vents locations are distributed over a range spanning approximately 40 feet from East to West and approximately 25 feet from North to South. The release elevation of the MSSV vents is slightly lower than the release elevation of the ARV vents. Choosing the release point nearest to the receptor entailed selecting between release from an MSSV vent or release from an ARV vent. For the paths to the Emergency CR Air Intake and the Normal CR Air Intake, the nearest release point corresponded to release from an MSSV vent. For the path to the TSC Air Intake, the nearest point corresponded to release from an ARV vent. Thus, the reported release heights are slightly different. Assumption 3 in section 4.1.2.3 "Assumptions and Acceptance Criteria" of the LAR specifically addresses this aspect of the MSSV/ARV Vent release path:

> "In case of release from MSSV/ARV vents, the release point exists at the point nearest to the intake vent that exists on the perimeter of the smallest rectangle that encloses all MSSV/ARV vents. This yields a conservatively short distance between the MSSV/ARV release point and the receptor. Because the MSSV and ARV vents have different discharge elevations, this results in different release heights being used for the control room and TSC calculations."

Turbine-Driven AFW exhaust

For releases from the Turbine-Driven AFW exhaust, the discrepancy in the release height is about 7 inches. The slant distance between the source and the receptor locations can be calculated using a release point either at the centerline or at the periphery of the Turbine-Driven AFW exhaust pipe. The choice of choosing either the centerline or the periphery of an exhaust pipe as a release point can lead to different release heights. However, it is the slant distance that is used to calculate X/Q in ARCON96, and the slant distance is internally calculated from the input values of release height and horizontal distance. For this particular case, the input value for the horizontal distance was not independently determined. It was adjusted based on the choice of the release point. First, the total shortest slant distance was determined beforehand. The shortest slant distance was then used to calculate the horizontal distance based on the chosen release height so that the "adjusted horizontal distance" is used as an input. Hence, a combination of release height input and "adjusted horizontal distance" input will always yield the shortest slant distance.

For the Turbine-Driven AFW exhaust calculations to the Emergency CR Air Intake and TSC Air Intake, the release elevation was selected to be the centerline of the Turbine-Driven AFW exhaust pipe. For the Normal CR Air Intake calculation, the release elevation was selected to be the top of the Turbine-Driven AFW exhaust pipe. However, due to the nature of the distance

calculation, the difference in release elevation has no impact on the resulting atmospheric dispersion factors. All three calculations appropriately calculate the shortest slant distance between the source and the receptor location.

Radwaste Building

For releases from the Radwaste Building, it was assumed that releases could originate from any point on the exterior surface of the Radwaste Building. For the Emergency CR Air Intake and TSC Air Intake receptor locations, the shortest path originates from a point on the Radwaste Building that is at the same elevation as the receptor elevation. The shortest path from the Radwaste Building to the Normal CR Air Intake receptor location travels over the roof of the Auxiliary Building. Because of this, the nearest point on the Radwaste Building is at the top of the building.

References:

1. WCNOC Letter ET 17-0001, "License Amendment Request for the Transition to Westinghouse Core Design and Safety Analyses," January 17, 2017. ADAMS Accession No. ML17053B393.

2. Electronic mail from B. K. Singal, USNRC, to W. T. Muilenburg, WCNOC, "Request for Additional Information - License Amendment Request for Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term Wolf Creek Generating Station (CAC No. MF 9307)," June 14, 2017. ADAMS Accession No. ML17166A038.

3. Westinghouse NSAL-00-016, "Rod Withdrawal from Subcritical Protection in Lower Modes," December 4, 2000.

4. Letter from K. D. Young, Ameren Missouri, to USNRC, "Callaway, Application for Amendment, Proposed Amendment Would Add New Technical Specification (TS) LCO 3.1.9, 'RCS Boron Limitations < 500F,' and Revise TS 3.3.1, 'Reactor Trip System (RTS) Instrumentation,' " April 14, 2005. ADAMS Accession Number ML051120189.

5. Letter from J. N. Donohew, USNRC, to C. D. Naslund, Ameren Missouri, "Callaway Plant, Unit 1 - Issuance of Amendment RE: Reactor Coolant System Boron Limitations in Modes 3, 4, and 5 (TAC NO. MC6897)," August 21, 2006. ADAMS Accession Number ML061590406.