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

April 3, 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: TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSIS (TAC NO. MF2574)

Dear Mr. Heflin:

By application dated August 13, 2013, to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System (ADAMS) package Accession No. ML13247A075), Wolf Creek Nuclear Operating Corporation (the licensee) requested a license amendment for Wolf Creek Generating Station, to revise the Technical Specifications to support transition to the Westinghouse core design and safety analysis.

The NRC staff has reviewed the information provided in your application and determined that additional information is required in order to complete its review. The enclosed questions were provided to Mr. S. Wideman of your staff on March 21, 2013. Please provide a response to the questions within 90 days of the date of this letter.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of NRC staff resources. If circumstances result in the need to revise the requested response date, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

Balwant K Singal for

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

TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSIS

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

By application dated August 13, 2013, to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System (ADAMS) package Accession No. ML13247A075), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) requested a license amendment for Wolf Creek Generating Station (WCGS), to revise the Technical Specifications (TSs) to support transition to the Westinghouse core design and safety analysis. The proposed amendment would revise the WCGS Operating License by incorporating a full scope implementation of the Alternative Source Term (AST) radiological analysis in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67 (10 CFR 50.67), "Accident Source Term," revising TS definitions and adopting Technical Specification Task Force Traveler 51 (TSTF-51), Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations," dated November 1, 1999 (ADAMS Accession No. ML993190284).

The requirements of 10 CFR 50.67 provide that the applicant's analysis must demonstrate that certain onsite and control room dose limits are met in order for a license amendment application to be approved. In order for the NRC staff to determine if the 10 CFR 50.67 criteria are met, the following additional information, as listed below, is required for the staff to complete its review.

Please note that many of the following requests for additional information (RAIs) might be answered by information already contained in the radiological accident analysis calculations performed by the licensee. Where the information is already provided in the calculations, it is acceptable to provide the calculations and state where the information is located. It is helpful to provide these calculations to the NRC staff, because it usually increases the efficiency of the review.

Note that all RAIs, unless otherwise specified, refer to the August 13, 2013, submittal. For example, ARCB-RAI-2 states: "Enclosure VI, page 4-38 states …." This should be inferred to be Enclosure VI (ADAMS Accession No. ML13247A080), page 4-38 of the August 13, 2013, submittal.

ARCB-RAI-1

Please justify all changes from the current licensing basis (see Issue 1 of NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347), for more detail). No justification is needed for changes that are consistent with Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS

Accession No. ML003716792), or are provided in the submittal dated August 13, 2013 (ADAMS Accession No. ML13247A076), unless requested by these RAIs.

ARCB-RAI-2

Enclosure VI, page 4-38 states, in part, that

> An additional fuel management multiplier is applied to the calculated core activity to account for anticipated variations in fuel cycle design.

Please provide the value of the multiplier, how it was derived, and a justification for its use.

ARCB-RAI-3

Enclosure VI, page 1-1 proposes an exception to the full scope implementation of the AST methodology. The "exception" to retain the current licensing basis (using Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (not publicly available, proprietary information), for NUREG-0737, "Clarification of TMI Action Plan Requirements," July 2000 (ADAMS Accession No. ML051400209), evaluations other than Control Room Habitability Envelope (CRHE) and Technical Support Center (TSC) doses) is based upon NRC Regulatory Guide (RG) 1.183, Section 1.3.5, "Equipment Environmental Qualification" and Section 6, "Assumptions for Evaluating the Radiation Doses for Equipment Qualification." These RG 1.183 sections are only for equipment qualification and are not for NUREG-0737 evaluations.

RG 1.183, Regulatory Position 4.3, under "Other Dose Consequences," states, in part, that

> The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737.... Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE [total effective dose equivalent].

a) Please confirm if the current licensing basis calculations are used to determine post-accident vital area access unaffected or bounding (using the TID-14844 source term) compared to those using the AST.

b) Please explain if any new operator actions have been credited to support the AST methodology.

c) Please explain if the current licensing basis calculations are used to determine the doses for operation of the post-accident sampling system or the containment high range radiation monitors used to monitor post-accident primary containment radiation levels (using the TID-14844 source term) unaffected or bounding compared to those using the AST.

d) Please provide a detailed justification for the proposed exemption or follow Regulatory Position 4.3 of RG 1.183.

ARCB-RAI-4

Enclosure VI, page 5-2 states, in part, that

> Core design parameters (enrichment, burnup, and MTU loading) are based on cycle 19 core design.

a) Please describe how many batches or core regions were assumed to determine the source term.

b) Please explain what period of irradiation (burnup) and specific power (i.e., megawatts days per metric ton of uranium (MWD/MTU)) was assumed for each region.

c) Please explain if the activity from each batch was taken at the end of life for each cycle.

d) Please provide the number of dose significant isotopes form the source term input for the RADTRAD dose evaluations.

e) For the purpose of the design basis, please explain the maximum enrichment assumed and if the assumed period of irradiation allows for the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.

ARCB-RAI-5

Enclosure VI, page 15B-3 states that the analysis conforms to Regulatory Position 3.2 of RG 1.183. Please confirm that reactor fuel will have a peak burnup of less than 62,000 MWD/MTU of uranium and a maximum linear heat generation rate of 6.3 kilowatts per foot (kW/ft) or less for peak rod average power for burnups exceeding 54,000 MWD/MTU. If not, please justify how the analysis conforms to Regulatory Position 3.2.

ARCB-RAI-6

Enclosure VI, page 4-63 states, in part, that

> Sedimentation is credited in the portion of containment that is not impacted by spray removal and in the sprayed portion when sprays are not on at a rate of 0.1 hr^{-1} until a DF [decontamination factor] of 1000 is reached at 23.5 hours. After this time sedimentation removal is terminated.

Removal of aerosol by sprays and natural deposition are competing processes. Please justify crediting both spray removal and the proposed 0.1 hr^{-1} sedimentation rate. Describe how the natural deposition model accounts for removal due to the spray model used. If any further credit for a reduction in aerosols is taken for any pathway, please provide a justification for that credit

while considering the impact of any other removal mechanism credited. Please confirm if the assumed sedimentation rate impacts the final dose.

ARCB-RAI-7

Enclosure VI, page 4-63 states, in part, that

> The resulting removal coefficient for elemental iodine is 22.9 hr^{-1}. SRP [Standard Review Plan] 6.5.2 allows for elemental iodine removal credit of up to 20 hr^{-1} during injection spray; however, to avoid sensitivities with spray switchover times from injection to recirculation and to conservatively address iodine loading in the spray fluid during recirculation, the removal is limited to 10 hr^{-1} for either spray mode.

Please explain in more detail why the removal coefficient is determined to be limited to 10 hr^{-1} and what is meant by "sensitivities with spray switchover times from injection to recirculation and to conservatively address iodine loading in the spray fluid during recirculation."

ARCB-RAI-8

Enclosure VI, page 4-64 states, in part, that

> An adjustment is made to account for a reduction in the RWST [refueling water storage tank] gas volume available for dilution as the leakage into the RWST increase the water level.

Please provide details regarding this adjustment so that the NRC staff can independently confirm the doses from the RWST back-leakage.

ARCB-RAI-9

Please provide the doses from each pathway analyzed for the loss-of-coolant accident (for the exclusion area boundary (EAB) (worst 2 hours), low population zone (LPZ), and control room (at 30 days)). Please explain whether the worst 2 hours EAB dose is determined using the sum of the worst 2 hours dose for each pathway to the EAB or by first summing all the time dependent dose pathways and then determining the worst 2 hour dose.

ARCB-RAI-10

Enclosure VI, page 4-38 states, in part, that

> The FIPCO-V computer code calculates the buildup of fission product activities in plant systems and components, including the reactor coolant system, chemical and volume control system demineralizer resins, VCT [volume control tank] liquid and vapor phases, and waste decay tank (WGDT).

Please provide details regarding the input and methodologies used in the FIPCO-V code for the staff to replicate the calculations performed by the FIPCO-V code.

ARCB-RAI-11

For the containment purge pathway analyzed for the loss-of-coolant accident (LOCA), please describe what is the assumed form of radioiodine released from the reactor coolant system prior to isolation of the containment purge. Please justify your answer.

ARCB-RAI-12

NRC Information Notice (IN) 2012-01, "Seismic Considerations - Principally Issues Involving Tanks," dated January 26, 2012 (ADAMS Accession No. ML11292A175), provides examples and references to events in which licensees failed to recognize various seismic considerations and system alignment issues that could impact safety. The NRC staff has identified recent concerns about licensees who failed to recognize that aligning non-seismic piping to the RWST would require TS limiting condition for operation (LCO) action statement entry, system modifications, or license amendments.

RG 1.183, Regulatory Position 5.1.2, "Credit for Engineered Safeguard Features," states:

> Credit may be taken for accident mitigation features that are classified as safety related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

During operations at WCGS, please describe if there is any non-seismic piping aligned to the RWST or systems which recirculate post-LOCA sump fluid. Also, provide details of how the AST analysis is modeled. Please provide enough detail so the NRC staff can independently model this configuration to assess its impact on design basis accident (DBA) doses.

ARCB-RAI-13

According to Enclosure VI, Table 4.3-5, "Control Room and Control Room Building Parameters," the delay due to switch to emergency mode operation following receipt of isolation signal in the current licensing basis is stated as "N/A" or not applicable. The submittal proposes to change this delay to 60 seconds. An isolation set point for the R-23 detector is also provided.

Tables are provided for each accident with analysis parameters and assumptions (i.e., Tables 4.3-6 through Table 4.3-16.) These tables include either the time "delay to switch to emergency mode of operation after event initiation," or the time "delay to switch to emergency mode operation following receipt of isolation signal."

a) For each change to the current licensing bases, please provide details about how the total time to isolate was calculated.

b) Please state and justify the isolation signal assumed to initiate the switch to emergency mode.

c) Please state whether the time to isolate includes the worst case single failure, control room isolation signal time, emergency diesel generator startup time, time to load the electrical bus, and damper closure time. If not, please justify why these assumptions and delays are not considered.

d) Please describe how the R-23 detector is used to isolate the control room and for which accidents it is credited. Provide enough detail so that the NRC staff can independently calculate the isolation time credited.

e) Please state if the R-23 detector is a general area monitor or is it located in the control room HVAC ductwork.

f) Please state if the R-23 detector and initiation signal comply with Regulatory Position 5.1.2, "Credit for Engineered Safeguard Features" of RG 1.183 and explain how compliance with Regulatory Position 5.1.2 is met.

ARCB-RAI-14

Please provide a diagram describing the model used for modeling the control building and control room for each design basis accident. Please provide the unfiltered in-leakage into the control building and justify the value. Please explain if this value has been confirmed by testing and if it will be confirmed periodically as part of a TS surveillance program. (Note this issue was previously identified in NRC Regulatory Issue Summary 2006-04, Issue Number 3.)

ARCB-RAI-15

In the proposed TS Bases B 3.3.8, "Emergency Exhaust System (EES) Actuation Instrumentation," which ensures that radioactive materials in the fuel building atmosphere are filtered and absorbed prior to exhausting to the environment, the EES is not credited after 76 hours of decay from the fuel. Without the EES credited, the leakage (source) could occur anywhere there is a penetration or hole in the fuel building (rather than through the exhaust of the EES). Please explain how the worst-case source-receptor pairings are determined for the calculation of atmospheric dispersion factors when the EES is not credited.

ARCB-RAI-16

WCNOC assumes the control room does not isolate after a fuel handling accident. The normal unfiltered outside air makeup flow to the control building and the control room is 13,050 cubic feet per minute (cfm) and 1950 cfm, respectively. The unfiltered in-leakage to the control room is assumed to be 50 cfm.

a) Please justify the use of 50 cfm for unfiltered in-leakage in the configuration when the control room ventilation system is not isolated. Please state if this value has been confirmed by testing and will it be confirmed periodically as part of a TS

surveillance program. (Note this issue was previously identified in NRC Regulatory Issue Summary 2006-04, Issue Number 3.)

b) In this case only, the normal makeup for the control building and control room is used to mitigate the consequences of the fuel handling accident. Please state if the normal control room heating and ventilation systems is credited meet the qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. (Note this issue was previously identified in NRC Regulatory Issue Summary 2006-04, Issue Number 3.) If so, please justify how the credited ventilation system complies with Regulatory Positions 4.2.4 and 5.1.2. If not, justify the proposed alternative assumptions used.

c) Please state if there is a surveillance for these normal makeup flow rates in the TSs. If not, please explain if a sensitivity analysis has been performed to determine the limiting dose based upon the range of possible makeup flow rates. Please provide the results (dose versus makeup flow rates) of any sensitivity analyses performed.

d) For other accidents that do not assume control room isolation is actuated (i.e., locked reactor coolant pump rotor, loss of alternating current power, letdown line break and tank ruptures) please explain if the assumptions used to model the control building or control room are the same as those used for the fuel handling accident. In your submittal for these accidents, where the control room isolation is not credited, the control room ventilation system is assumed to remain in the normal mode of operation. Please state whether this assumption yields more conservative doses than if the emergency mode is assumed to be actuated and justify your answer. For these accidents (other than the fuel handling accident) that do not credit control room isolation, please explain if a sensitivity analysis has been performed to determine the limiting dose based upon a range of possible makeup flows. If so, please provide the results of the analysis (dose vs. makeup flow). If not, justify why the assumed makeup flows are limiting.

ARCB-RAI-17

The current licensing basis for the radioactive waste gas decay tank failure, from Updated Final Safety Analysis Report (UFSAR) Section 15.7.1.2, states that the tank is assumed to fail after 40 years, releasing the peak inventory expected in the tank. The proposed change requests a change to this assumption. Please justify this change and explain why it is conservative.

ARCB-RAI-18

Page 15.7-13 of the UFSAR "markup" states that the gap fractions are obtained for high burnup fuel from Regulatory Guide (RG) 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," dated March 1972 (ADAMS Accession No. ML083300022), as modified by NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," dated February 1988 (not publicly available), to support the conservative assumption that 100 percent of the rods do not

meet the burnup and kilowatts per foot (kW/ft) limits set forth in Footnote 11 of RG 1.183. Enclosure VI, page 4-69 of the submittal elaborates on the use of NUREG/CR-5009.

a) Please justify the use and applicability of NUREG/CR-5009 instead of industry standards such as ANSI/ANS-5.4-2011, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel."

b) Please justify the statement made on page 3A-12 of the UFSAR markup that "Use of this regulatory guide [RG 1.25] has been replaced by Regulatory Guide 1.183 for alternative source term application," in light of the statement that the gap fractions are obtained from RG 1.25.

ARCB-RAI-19

The current single failure taken for the fuel handling accident is the failure of the humidly control system for the engineered safety feature (ESF) emergency filtration system as stated on page 15.7-13 of the UFSAR. Please explain the worst case single failure that is assumed for the fuel handling accident in the proposed analysis. Please justify your answer.

ARCB-RAI-20

Enclosure VI, page 4-69 of the submittal states, in part, that

> Although not explicitly discussed, the specified overall DF [decontamination factor] also applies to rod internal pressures up to 1500 psig [pounds per square inch gauge].

The DF of 200 provided in RG 1.183 is based upon Reference B-1 ("Evaluation of Fission Product Release and Transport," dated October 5, 1971 (ADAMS Legacy Accession No. 8402080322)) of RG 1.183. The data upon which the pool DF of 200 is based was developed in 1971 and was based on the Westinghouse fuel marketed at the time (the assumed internal fuel pressure of 1200 psig was used). Since higher pressures correlate to lower DFs, the NRC staff is concerned that a DF of 200 might not be sufficiently conservative for pressures higher than 1200 psig.

Please provide the data for current fuel types used at WCNOC that justify a DF of 200 for fuel pressures up to 1500 psig. Also, please provide a detailed justification for using a DF of 200 for pressures up to 1500 psig.

ARCB-RAI-21

Enclosure VI, page 4-69 of the submittal states, in part, that

> The decay time used in determining the inventory of the damaged rods is 76 hours. Thus, the analysis supports the **TS limit** [emphasis added] of 76 hours decay time prior to fuel movement.

The decay time appears to meet Criterion 2 of 10 CFR 50.36, "Technical specifications," but the NRC staff could not locate the TS limit of 76 hours, though the limit of 76 hours is in the proposed TS Bases and FSAR pages. Please identify where in the TSs this limit exists, or justify why a TS limit is not being proposed.

ARCB-RAI-22

Please confirm that with the exception of different release points, the assumptions and inputs are identical for the fuel handling accident within the containment and the fuel handling accident outside the containment.

ARCB-RAI-23

a) Please confirm that the most limiting combination of release point and receptor for the control room and technical support center (TSC) were used to determine atmospheric dispersion factors for each accident.

b) Please explain why the control room atmospheric dispersion factors provided in the tables for individual accidents (i.e., Tables 4.3-11 and 4.3-15) do not correlate to values provide in Table 4.1.2-3.

c) Please state and justify the release points that correlate to the atmospheric dispersion factors for each design basis accident.

ARCB-RAI-24

In a letter dated November 7, 2013 (ADAMS Accession No. ML13246A358), the NRC informed the Technical Specifications Task Force of concerns that the NRC staff had recently identified during a review of plant-specific license amendments requesting adoption of three travelers including traveler TSTF-51, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations."

Enclosure VI, page 2-4 of the submittal discusses the proposed TSTF-51 changes.

TSTF-51 states, in part, that

> The addition of the term "recently" associated with handling irradiated fuel in all of the containment function Technical Specification requirements is only applicable to those licensees who have demonstrated by analysis that after sufficient radioactive decay has occurred, off-site doses resulting from a fuel handling accident remain below the Standard Review Plan limits (well within 10CFR100). [or 10 CFR 50.67]

NUREG-0800, Standard Review Plan (SRP) 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," July 2000 (ADAMS Accession No. ML003734190), states, in part, that

> The models, assumptions, and parameter inputs used by the licensee should be reviewed to ensure that the conservative design basis assumptions outlined in RG-1.183 have been incorporated.

Appendix B of RG 1.183, Regulatory Position 1.1 states:

> The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or weight of a dropped fuel assembly...

With regard to the WCGS submittal to adopt TSTF-51, please provide plant-specific information to verify that the limiting cases have been considered, e.g., a fuel handling accident analysis that evaluates the dropping of loads allowed over irradiated fuel assemblies (i.e., new fuel assembly, sources, or reactivity control components) onto irradiated fuel assemblies prior to and after the proposed 76-hour decay time. Such an analysis should only credit those safety systems required to be operable as required by TS. There must be reasonable assurance that the fuel handling accident analysis doses remain within regulatory limits when references to Core Alterations are removed from TSs and Engineered Safety Features are no longer required during movement of loads such as new fuel assemblies, sources or reactivity control components.

ARCB-RAI-25

Several calculations assume an iodine partition in the steam generators of 100 which is applied to the releases resulting from steaming of secondary side fluid (i.e., Enclosure VI, page 4-59). Please confirm that the partition factor is applied only to the elemental iodine or justify how organic iodine is partitioned.

ARCB-RAI-26

The column labeled "Comments" of Enclosure VI, Table A for Regulatory Position 4.1.1, states that progeny was not included in the dose calculations consistent with the two previously approved submittals. The NRC staff's review of the subject safety evaluations of these two submittals did not find explicit approval or a review of excluding the effects of progeny. Since this appears to conflict with Regulatory Position 4.1.1 and could potentially yield a non-conservative estimate of doses, please either provide justification for not including the progeny, or include progeny in these calculations.

ARCB-RAI-27

For those accidents that model accident and pre-existing spikes, RG 1.183 specifies the release from the fuel to be in the form of 95 percent cesium iodide. Please confirm that the spike

modeled in the reactor coolant system models the increase in cesium in addition to the iodine modeled.

ARCB-RAI-28

The column labeled "Comments" of Enclosure VI, Table F for Regulatory Position 5.6 and Table G for Regulatory Position 7.4 states that: "The transport model described in Regulatory Positions 5.5 and 5.6 ... was considered as appropriate [emphasis added] ..." Please clarify if the models described in these Regulatory Positions were used. If only those considered appropriate were used, please state which were not considered appropriate and justify why they are not appropriate.

ARCB-RAI-29

Enclosure VI, page 4-54 states, in part, that

> The minimum SG [steam generator] water mass is increased after 2 hours to take credit for operators maintaining level at narrow range just on span.

Please explain if this this assumption was credited previously. If not, please provide a justification for this assumption.

ARCB-RAI-30

Regulatory Position 5.1.2 of RG 1.183 states, in part, that

> The single active component failure that results in the most limiting radiological consequences should be assumed.

Please provide the most limiting single active failure for each design basis accident.

ARCB-RAI-31

Enclosure VI, page 15.7-1 of the UFSAR markups indicates a proposed change in the licensing bases for the waste gas decay tank failure. Previously, the tank was assumed to fail after 40 years, releasing the peak inventory expected in the tank. The proposed change is assumed to be the maximum activity for each radionuclide during the degassing operations and the Krypton-85 inventory is assumed to be the total activity released during the fuel cycle. RG 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure," March 1972 (ADAMS Accession No. ML083300020), states, in part, that

> The maximum content [emphasis added] of the decay tank assumed to fail should be used for the purpose of computing the noble gas inventory in the tank

a) If previously the tank was assumed to accumulate activity over 40 years, please explain why the activity from one fuel cycle is conservative and consistent with the RG 1.24 above regulatory position of using the maximum content of the

decay tank. Please justify the proposed change in assumptions regarding the tank contents and state why they are conservative.

b) Table 4.3-2a contains 3 columns of source terms used to calculate the tank failures described in Enclosure VI, Sections 4.3.10, "Waste Gas Decay Tank Failure (USAR [UFSAR] Chapter 15.7.1.5)," and 4.3.11, "Liquid Waste Tank Failure (USAR [UFSAR] Chapter 15.7.2.5)." Please state the assumptions used to calculate these source terms.

ARCB-RAI-32

Enclosure VI markups of UFSAR page 15.3-11 change the sentence from "Steam generator tube leakage is assumed to continue until the pressures in the reactor coolant and secondary systems are equalized" to "Steam generator tube leakage is assumed to continue until the residual heat removal system can match decay heat and release from the secondary system are terminated." Please justify this proposed change.

ARCB-RAI-33

Enclosure VI markups of UFSAR page 6.5A-3 add an Insert N which defines the term k_g or the "gas-phase mass-transfer coefficient." The k_g used is taken from Reference 2 or Brookhaven National Laboratory (BNL) Technical Report A-3788, "Fission Product Removal Effectiveness of Chemical Additives in PWR Containment Sprays," August 1986, rather than using the reference (Reference 20, entitled: "The Terminal Speed of Single Drops or Bubbles in an Infinite Medium," *International Journal of Multiphase Flow*, pages 491-511 (1974)) cited in SRP 6.5.2, "Containment Spray as a Fission Product Cleanup System." Justify why Reference 2 is appropriate rather than the study cited in the SRP and state why it was used. Please provide a copy of the BNL report.

ARCB-RAI-34

Enclosure VI, page 8-14 states in the comments for NRC Regulatory Issue Summary 2006-04, Issue 6 that there were no changes to the plant configuration. The changes to TS 3.9.4 allow an "open" containment when moving fuel that is not recently irradiated. Consistent with Regulatory Issue Summary 2006-04, please confirm that all pathways to the environment created by the proposed changes are considered and analyzed in the fuel handling accident analysis.

ARCB-RAI-35

Enclosure VI, markups to UFSAR Table 15.6-4 remove the words "with forced overfill" from the steam generator tube rupture. Please explain why this "forced overfill" was part of the licensing bases and what has changed to allow the change to no longer consider forced overfill.

ARCB-RAI-36

The column labeled "Comments" of Enclosure VI, Table D for Regulatory Position 5.6 discusses WCAP-13247 and an NRC staff response dated March 10, 1993. The NRC staff is unable to locate these documents. Please provide these two documents.

ARCB-RAI-37

Please provide a complete description of how the dose equivalent iodine-131 (DE) and the dose equivalent Xenon-133 are calculated.

ARCB-RAI-38

SRP 16.0 states, in part, that

> In TS change requests for facilities with TS based on previous STS [Standard Technical Specifications], licensees should comply with comparable provisions in these STS NUREGs to the extent possible or justify deviations from the STS.

Several of the proposed changes to the TSs and TS bases (an example is provided below) do not align with the STS. Please provide a justification for deviations from the STS.

For example, proposed page B 3.6.3-2 removes detail from the bases of TS 3.6.3, "Containment Isolation Valves," which is inconsistent with the STS, NUREG 1431, and Revision 4. The detail provides the limiting total response time for the isolation of containment. Please provide the value used in the DBA analyses assumed for containment isolation.

April 3, 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: TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSIS (TAC NO. MF2574)

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The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of NRC staff resources. If circumstances result in the need to revise the requested response date, please contact me at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

/RA by BSingal for/

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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RidsNrrPMWolfCreek Resource
RidsRgn4MailCenter Resource
MBlumberg, NRR/DRA/ARCB

ADAMS Accession No.: ML14083A400 ***memo dated March 20, 2014**

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DRA/ARCB/BC*	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME	FLyon	JBurkhardt	UShoop	MMarkley	FLyon (BSingal for)
DATE	04/02/14	03/25/14	03/20/14	04/03/14	04/03/14

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