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

December 4, 2017

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 FOR TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSES INCLUDING ADOPTION OF ALTERNATIVE SOURCE TERM (CAC NO. MF9307; EPID L-2017-LLA-0211)

Dear Mr. Heflin:

By letter dated January 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17054C103), as supplemented by letter dated May 4, 2017 (ADAMS Accession No. ML17130A915), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee), submitted a license amendment request (LAR) for Wolf Creek Generating Station (WCGS). The proposed amendment would revise the WCGS Technical Specifications and the Updated Safety Analysis Report (USAR) 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." The U. S. Nuclear Regulatory Commission (NRC) staff reviewed the impact of implementing an alternative radiological source term for evaluating the currently analyzed design-basis accidents in the WCGS USAR and identified the enclosed additional information is needed in order to complete the review of the LAR.

The NRC staff issued a draft request for additional information (RAI) on November 3, 2017. An RAI clarification call was held on November 21, 2017. It was agreed that the licensee will provide a response to all of the RAIs by January 11, 2018.

If you have any questions, please contact me at 301-415-3016 or via e-mail at Balwant.Singal@nrc.gov.

Sincerely,

Balwant K. Singal, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosure
Request for Additional Information

cc: Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO TRANSITION

TO WESTINGHOUSE CORE AND DESIGN ANALYSES

INCLUDING ADOPTION OF ALTERNATIVE SOURCE TERM

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

By letter dated January 17, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17054C103), as supplemented by letter dated May 4, 2017 (ADAMS Accession No. ML17130A915), Wolf Creek Nuclear Operating Corporation (WCNOC, the licensee) submitted a license amendment request (LAR) for Wolf Creek Generating Station (WCGS). The proposed amendment would revise WCGS Technical Specifications (TSs) and the Updated Safety Analysis Report (USAR) Chapter 15, "Accident Analysis," 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." The U. S. Nuclear Regulatory Commission (NRC) staff reviewed the impact of implementing an alternative radiological source term for evaluating the currently analyzed design-basis accidents (DBAs) in the WCGS USAR and identified the following additional information is needed in order to complete the review of the LAR.

Please note that the request for additional information (RAI) represents input from the following technical review branches:

- Radiation Protection & Consequence Branch (ARCB)
- Probabilistic Risk Assessment Operations & Human Factors Branch (APHB)
- Technical Specifications Branch (STSB)

The NRC staff issued draft RAIs on November 3, 2017. An RAI clarification call was held on November 21, 2017. It was agreed that the licensee will provide response to all of the RAIs by January 11, 2018.

Radiation Protection & Consequence Branch (ARCB)

Regulatory Analysis Basis

1) Section 50.67 of 10 CFR allows licensees, seeking to revise their current accident source term in design-basis radiological consequence analyses, to apply for a license amendment under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit." The application shall contain an evaluation of the consequences of applicable DBAs previously analyzed in the safety analysis report. Paragraph 50.67(b)(2) of 10 CFR requires that the licensee's analysis demonstrates with reasonable assurance that:

    (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated

fission product release, would not receive a radiation dose in excess of 0.25 Sv [sievert] (25 rem) [roentgen equivalent man] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

2) Part 20 of 10 CFR, "Standards for Protection Against Radiation," Subpart C, "Occupational Dose Limits," Section 20.1201, states, in part:

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures under § 20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of—

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or
(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin of the whole body, and to the skin of the extremities, which are:

(i) A lens dose equivalent of 15 rems (0.15 Sv), and
(ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

(b) Doses received in excess of the annual limits, including doses received during accidents, emergencies, and planned special exposures, must be subtracted from the limits for planned special exposures that the individual may receive during the current year (see § 20.1206(e)(1)) and during the individual's lifetime (see § 20.1206(e)(2)).

3) Part 20 of 10 CFR, Subpart D, "Radiation Dose Limits for Individual Members of the Public," Section 20.1301(a), states, in part:

(a) Each licensee shall conduct operations so that –

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv [millisievert])

in a year, exclusive of the dose contributions from background radiation, from any medical administration the individual has received, from exposure to individuals administered radioactive material and released under § 35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003, and

(2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with § 35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

4) Subpart D to 10 CFR 20.1301(e) states:

(e) In addition to the requirements of this part, a licensee subject to the provisions of EPA's [Environmental Protection Agency's] generally applicable environmental radiation standards in 40 CFR part 190 shall comply with those standards.

5) Part 190 of 40 CFR, "Environmental Radiation Protection Standards for Nuclear Power Operations," Section 190.10, "Standard for normal operations," states, in part:

Operations covered by this subpart shall be conducted in such a manner as to provide reasonable assurance that:

(a) The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations.

6) NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition (SRP), Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190).

7) Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

8) Section 50.90 of 10 CFR, states, in part, that whenever a holder of an operating license under this part, desires to amend the license, application for an amendment must be filed with the Commission, as specified in Section 50.4 of this chapter, as applicable, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

9) Appendix A to 10 CFR Part 50, "General Design Criteria [GDC] for Nuclear Power Plants": Criterion 19, "Control room," requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. It also states that adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

10) Section 50.36, "Technical specifications," of 10 CFR contains the NRC's regulatory requirements related to the content of the TSs. This regulation requires that the TS include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notifications; and (8) written reports.

11) NUREG-0800, Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 4, March 2007 (ADAMS Accession No. ML070190178).

12) RG 1.194, "Revision 0, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003 (ADAMS Accession No. ML031530505).

13) RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003 (ADAMS Accession No. ML031490664).

RAI ARCB1-LOAC-1 - Loss of Non-Emergency Alternating Current Power (LOAC)

In Enclosure IV to the letter dated January 17, 2017 (Enclosure IV) (ADAMS Accession No. ML17054C227), WCNOC stated, in part:

> No fuel cladding damage or fuel melting is assumed to occur as a result of this [LOAC] accident.

The licensee further stated, in part:

> ...the release pathway for this analysis is similar to the locked rotor event and the accident-initiated iodine spike is similar to the MSLB [main steam line break] event. Therefore, release pathway models consistent with RG 1.183, Appendix G, and accident-initiated iodine spiking models consistent with RG 1.183, Appendix E, are applied to this analysis.

RG 1.183, Appendix E, Assumptions for Evaluating the Radiological Consequences of a PWR [Pressurized-Water Reactor] Main Steam Line Break Accident," Regulatory Position 2, states:

> 2. If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.

2.1 A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 µCi/gm [micro curie per gram] DE I-131 [dose equivalent iodine 131]) permitted by the technical specifications (i.e., a preaccident iodine spike case).

2.2 The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.

Enclosure IV, Section 4.3.4.2.1, "Source Term," states, in part:

WCGS's LOAC analysis assumes:

> The reactor trip associated with the LOAC creates an iodine spike that is assumed to increase the iodine release rate from the fuel to the RCS [reactor coolant system] to a value 500 times greater than the appearance rate corresponding to the maximum equilibrium RCS concentration of 1.0 µCi/gm DE I-131. ... The duration of the accident-initiated iodine spike is assumed to be 8 hours.

WCGS's source term assumption above appears to be consistent with RG 1.183, Appendix E, Regulatory Position 2.2, stated above. However, the NRC staff could not find the discussion or analysis of RG 1.183, Appendix E, Regulatory Position 2.1, for the LOAC event. Without this analysis the NRC staff does not have enough information to determine that WCGS's source term for the LOAC event is consistent with RG 1.183, Appendix E, as stated in Enclosure IV.

1. Please submit for the NRC staff's review an analysis or a description of the LOAC radiological consequences analysis assuming a pre-accident iodine spike. Please provide the inputs, assumptions, methodology technical basis for the analysis and the results of the analysis. Also, please justify the assumptions and inputs used in the analysis.

   or

2. Please explain how the LOAC analysis source term is consistent with the source term in RG 1.183, Appendix E, Regulatory Position 2.1.

RAI ARCB1-LOAC-2 - LOAC

Enclosure IV, Section 4.3.4, "Loss of Non-Emergency AC Power (USAR Section 15.2.6.3)," WCNOC states:

> 4.3.4.3, "Acceptance Criteria"
>
> The EAB [exclusion area boundary] and LPZ [low population zone] dose acceptance criterion for a LOAC is 0.1 rem TEDE, consistent with 10 CFR 20. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC [technical support center] dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 2 [Section 4.3.4.5].

The NRC staff needs clarity on the meaning of this paragraph. The first sentence seems to address dose limits for individual members of the public during license operations and the second sentence seems to address dose limits for an alternative source term (AST) which applies to accident analysis. These sentences are unclear to the NRC staff. Is WCNOC requesting that the acceptance criteria be (1) that under license operations 10 CFR 20.1201, 10 CFR 20.1301, and 40 CFR 190.10 <u>or</u> (2) that under accident criteria in 10 CFR 50.67 and RG 1.183? The NRC staff acknowledges that this accident isn't specifically addressed in an Appendix of RG 1.183. Common practice during application of the AST is for a licensee to choose the RG 1.183 Appendix that most closely matches the accident being analyzed. In this case it appears that the MSLB accident may bound the offsite releases for the LOAC and that the difference between the LOAC and the MSLB accident analyses is whether or not the control room and TSC ventilation systems are credited in emergency mode.

1. Please explain if WCNOC is requesting that the acceptance criteria be (1) that under license operations 10 CFR 20.1201, 10 CFR 20.1301, and 40 CFR 190.10 or (2) that under accident criteria in 10 CFR 50.67 and RG 1.183. In addition, provide the technical reasoning for the determination.

2. The NRC staff notes that WCNOC is required to comply with the regulations of 10 CFR Part 20 and after NRC approval of the AST, 10 CFR 50.67. This RAI is to determine which acceptance criteria is being requested in this license application by WCNOC.

RAI ARCB1-LLBA-1 (Letdown Line Break)

In Section 4.3.7, "Letdown Line Break [LLB] (USAR Section 15.6.2.1)," of Enclosure IV, WCNOC states, in part, in Section 4.3.7.2.2, "Release Model":

> It is assumed that 18% of the leaking coolant flashes to steam, based on the temperature and pressure conditions of the letdown line flow.

1. Please explain how the flashing fraction was determined. Was the flashing fraction determined consistent with RG 1.183, Appendix A, Regulatory Position 5.4 (which uses a constant enthalpy, h, process based on the maximum time-dependent temperature of the water circulating outside the containment: $FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$ where h_{f1} is the enthalpy of

liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation condition; and h_{fg} is the heat of vaporization at 212 degrees Fahrenheit).

RAI ARCB1-LLBA-2 (Letdown Line Break)

WCGS TS 3.4.16, "RCS Specific Activity" requires the RCS DE I-131 and dose equivalent XE-133 (DEX 133) specific activity shall be within limits." Specifically, iodine must be less than or equal to 1.0 micro curies per gram and, if exceeded, TS 3.4.16, Condition A, allows 48 hours to restore iodine to below the limit as long as DE I-131 is verified to remain equal to or below 60 µCi/gm. TS 3.4.16 assumes the initial reactor coolant iodine activity at 60 µCi/gm DE I-131 due to a pre-accident iodine spike caused by an RCS transient.

The LLB accident is an RCS transient that can cause a pre-accident iodine spike; therefore, the radiological consequences from this iodine spike must be evaluated for the LLB.

1. Please submit for the NRC staff's review an analysis or a description of the LLB accident radiological consequences analysis assuming a pre-accident iodine spike. Please provide the inputs, assumptions, methodology technical basis for the analysis and the results of the analysis (EAB, LPZ, control room and TSC). Also, please justify the assumptions and inputs used in the analysis.

RAI ARCB1-LLBA-3 (Letdown Line Break)

Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms" (ADAMS Accession No. ML053460347), states, in part:

> An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) <u>contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations</u>. The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable.

Enclosure IV, Section 8, "NRC Regulatory Issue Summary 2006-04 Comparison," WCNOC states that the submittal conforms to this RIS position. The Comments column states:

> The submittal is modeled after previous NRC-accepted submittals. Included is justification for each proposed change to the TS, identification of changes to licensing basis analyses, and sufficient analysis detail to allow for result verification through independent calculations.

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.

In addition, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," of Enclosure IV, states WCNOC's conformance with RG 1.183. It further states that only safety-related engineered safeguards features are credited in the analysis with an assumed single active failure that results in the greatest impact on the radiological consequences. A loss of offsite power is assumed concurrent with the start of each event as that maximizes the dose impact.

Enclosure IV, Section 4.1.2.4, "Results and Conclusions," WCNOC states, in part:

> For the letdown line break accident, the line break is assumed to occur outside of containment and radionuclides are directly released into the auxiliary building. The χ/Q [atmospheric dispersion factor] values for this case are those of the unit vent stack.

In WCGS USAR Section 15.6.2.1.1.4, "Identification of Leakage Pathways and Resultant Leakage Activity," WCNOC states, in part:

> However, the ventilation system is not designed to mitigate the consequences of an accident (e.g., it might not survive an earthquake more severe than the operating-basis earthquake), nor can the possibility of unplanned leakages from the auxiliary building be eliminated; hence, no credit is taken for these effects reducing the released activity.

1. Please state whether the auxiliary building or its ventilation systems are credited in the LLB analysis (for example for dilution, holdup or for the assumed point of release) or and other proposed design basis radiological consequences analysis. If so, please justify how these systems comply with RG 1.183, Regulatory Position 5.1.2.

<u>RAI ARCB1-LLBA-4</u> (Letdown Line Break)

RIS 2006-04, states, in part:

> An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations.

> The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable.

Enclosure IV, Section 8, "NRC Regulatory Issue Summary 2006-04 Comparison," WCNOC states that the submittal conforms to this RIS position and, the "Comments" column states:

> The submittal is modeled after previous NRC-accepted submittals. Included is justification for each proposed change to the TS, identification of changes to licensing basis analyses, and sufficient analysis detail to allow for result verification through independent calculations.

Enclosure IV, Section 4.3.7.2.2, "Release Model," WCNOC states, in part:

> Reactor coolant is assumed to be released at a rate of 141 gpm [gallons per minute] until the isolation valve is fully closed. The time required for the operator to identify the accident and close the letdown isolation valve is expected to be within 30 minutes, 10 seconds after accident initiation.

Table 4.3-10, "Assumptions Used for Letdown Line Break Analysis," of the letter dated May 4, 2017, states that the flow rate out the broken line is proposed to be 141 gpm and is not proposed to be changed from the current licensing basis.

Per Table 15.6-2 of the WCGS USAR, the current Letdown Line Break analysis assumes that the loss of reactor coolant is 444 gpm and the duration of the release [time to identify and close the letdown isolation valve] is 1810 seconds [30 minutes, 10 seconds].

The break flow is proposed to change (a decrease from 444 gpm to 141 gpm), but the time to identify the accident is assumed to remain the same. It would be expected that, if the break flow decreases, the time to identify the break flow could also decrease.

1. Please justify the new assumed break flow of 141 gpm and the time to identify the accident and close the letdown isolation. Please provide enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the values assumed.

RAI ARCB1-LOCA-1 - Loss-of-Coolant Accident (LOCA)

RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," Regulatory Position 3.2, states:

> Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1)[1] and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2)[1]. The latter model is incorporated into the analysis code RADTRAD (Ref. A-3)[1]. The prior practice of

[1] Reference number for SRP Section 6.5.2

> deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

Enclosure IV, Section 4.3.9.2.2.1, "Containment Leakage," states, in part:

> 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.

Per Enclosure IV, Table B, "Conformance with Regulatory Guide 1.183 Appendix A (Loss-of-Coolant-Accident)," the proposed LOCA analysis conforms with Regulatory Position 3.2 and the proposed aerosol deposition rate in the reactor containment for the unsprayed regions of containment (not accessible by containment sprays or when sprays are not operating) is 0.1 hr^{-1}. Other than the statement that this deposition rate was previously approved for Point Beach Nuclear Plant (ADAMS Accession No. ML110240054), no technical justification for its applicability to the WCGS design is provided.

The NRC staff's preliminary assessment of the proposed aerosol deposition rate in the unsprayed portions of containment indicates that using an aerosol deposition rate of 0.1 hr^{-1} (and restricting this use to unsprayed regions and to a DF limit of 1000) may be non-conservative. The sprays are very effective in removing aerosols and after they stop spraying the remaining aerosol in containment is small and would have high settling times.

1. Please explain how the removal coefficient(s) were calculated for the WCNOC design and how the assumptions are consistent with RG 1.183. Please provide enough detail (including the aerosol size distribution in containment after the sprays are stopped) to allow the NRC staff to confirm the methodology is conservative for the WCNOC design. Also, please provide the quantitative impact of the 0.1 hr^{-1} assumption on the dose results. Please note that NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (ADAMS Accession No. ML100130305), does not consider the impact of spray actuation.

RAI ARCB1-LOCA-2 - LOCA Control Room Modeling

Enclosure IV, Table 4.3-12 "Assumptions Used for LOCA Analysis," states that the "Time of control room isolation (including delays) (sec)" is given as 120 seconds.

1. Please confirm if this time delay is also assumed for the start of the control room recirculation filter. If not, justify why the delay is not included in the model of the control room.

RAI ARCB1-LOCA-3 - LOCA and other DBAs except Fuel Handling Accident (FHA)

Regarding the design basis LOCA analysis, Section 4.3.9.2.2.2, "Emergency Core Cooling System Leakage," of Enclosure IV, states, in part:

> The leakage to the auxiliary building is modeled at a rate of 2 gpm. The leakage value was doubled in accordance with RG 1.183. The analysis assumes that

> 10% of the iodine activity in the leakage becomes airborne and is available for release to the environment. The activity of the airborne leakage is further reduced as it is released through the auxiliary building vent filters with 90% efficiency for all forms of iodine.

RG 1.183, Regulatory Position 5.1.4, "Applicability of Prior Licensing Basis," states, in part:

> The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the AST analysis conforms to RG 1.183, Regulatory Position 5.1.4.

The Enclosure IV table entitled "Regulatory Guide 1.194 Comparison," states that the analysis in the LAR conforms to RG 1. 194. Regulatory Position 3.2.4.2 of RG 1.194 states:

> Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used. Releases from personnel air locks and equipment hatches exposed to the environment, or containment purge releases prior to containment isolation, may need to be treated differently. It may be necessary to consider several cases to ensure that the χ/Q value for the most limiting location is identified.

Section 50.36 of 10 CFR, requires the TSs to be derived from the analyses and evaluation included in the safety analysis report. Per WCGS TS Bases B 3.7.13, the Emergency Exhaust System (EES) design basis is established by the consequences of the limiting DBAs, which includes a LOCA.

A note in WCGS TS 3.7.13, "Emergency Exhaust System (EES)," allows the auxiliary building boundary to be opened intermittently under administrative controls and Condition B allows two EES trains to be inoperable due to an inoperable auxiliary building boundary for 24 hours during Modes 1-4 (when a LOCA could occur).

The proposed TS Bases, B 3.7.13, "Emergency Exhaust System (EES)," provided in the May 4, 2017, supplement, states, in part:

> Total system failure could result in the atmospheric release from the auxiliary building exceeding the guideline limits of 10 CFR 50.67 (Ref. 5)[2] in the event of a LOCA or fuel handling accident.

In Enclosure IV, Table A, the licensee states that the WCGS analysis conforms to RG 1.183, Regulatory Position 5.1.2, which states, in part, that credit may be taken for accident mitigation features that are required to be operable by TSs. Therefore, when the auxiliary building

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[2] 10 CFR 50.67, "Accident source term."

boundary is allowed to be inoperable (during Modes 1-4) it would not be credited in accident analysis used to establish the limiting DBA (for example, the LOCA).

However, the technical analysis section of the LAR does not include consideration for a scenario where the LOCA occurs while the auxiliary building boundary is open and not required to be restored, or the EES is not operable as allowed by the TS 3.7.13. For example, Enclosure VII to the letter dated January 17, 2017 (ADAMS Accession No. ML17054C229), Table 4.1.2-3(a), "Calculated χ/Q (sec/m3) for the Emergency Control Room Intake Vent," which includes the χ/Q values determined for the control room and TSC intakes, does not contain an atmospheric dispersion factor for a release from the auxiliary building boundary to the control room. Also, the LOCA analysis does not address potential direct pathways from the auxiliary building to the control room and the filtration from the auxiliary building vent filters are credited when the EES could be inoperable.

WCNOC has proposed to revise several design basis radiological analyses and the NRC staff must make a current finding of reasonable assurance that adequate protection will be maintained with the auxiliary building open and not required to be restored or the EES is inoperable and not credited (consistent with the TS discussed above). That assurance cannot be based solely on the probability of the accident not occurring since the applicable safety analysis for these TSs is based upon the fundamental assumption that the DBA occurs. The reasonable assurance assessment will include a determination of whether the limits in GDC 19 of Appendix A to 10 CFR Part 50 and 10 CFR 50.67 are met. In order for the NRC staff to make its assessment of reasonable assurance please provide the following information:

1. Submit for the NRC staff's review revised radiological consequences analyses of a LOCA (and any other design basis analyses other than the FHA). The analyses need to consider a scenario where the DBAs occur while the auxiliary and fuel building envelope boundaries (in addition to any other boundaries allowed to be open) are open for the duration of the accident, the EES is not credited and has dose results that meets the limits in GDC 19 of Appendix A to 10 CFR Part 50 and 10 CFR 50.67. In addition, provide the inputs, assumptions, methodology, and the results of the analysis. Also, please justify the assumptions and inputs in used in the analysis.

   or

2. Provide a proposed change to the Limiting Condition for Operation (LCO) 3.7.13 note so that it is consistent with proposed radiological consequence analyses and ensures that the auxiliary and fuel building boundaries are restored consistent with the shortest restoration time evaluated in the licensing basis analyses (including, as applicable, any consideration for obtaining the design basis pressures assumed in these analyses). Also, provide a proposed change to the completion time of LCO 3.7.13 Condition B to reflect the loss of safety function and unanalyzed condition, and take mitigating actions to ensure radiological exposures will not exceed the radiological limits in 10 CFR 50.67.

RAI ARCB1-LOCA-4 - LOCA

Enclosure IV, Section 4.4.2.6, "Results and Conclusions" states, in part:

> It should be noted that back-leakage of sump fluid through the RWST [refueling water storage tank] is not considered in the post-LOCA analysis based on the

plant emergency procedure for transfer to cold leg recirculation that requires closure of the 24" RWST outlet valve within 16 hours of SI [safety injection] initiation to limit releases from the RWST to the atmosphere. This results in three valve isolation and a minimum of two valve isolation in the long term with a single failure. Refer to USAR Table 6.3-5. Therefore, there is no need to address the potential issue of reduced pH in the RWST, which could lead to a potential radioactive iodine release from the RWST to the environment.

1. When the shift from the emergency core cooling system (ECCS) injection from the RWST to recirculation occurs, if the RWST still has borated (acidic) water left in it, there is a need to consider the pH in the RWST and the potential for additional radioactive releases from the RWST. If there is borated water in the RWST, please provide additional justification for the statement that there is no need to address the potential issue of reduced pH in the RWST, which could lead to a potential radioactive iodine release from the RWST to the environment.

2. Enclosure IV, Section 4.3.9.2.2.3, "Refueling Water Storage Tank Back-Leakage," states, in part:

> For the RWST back-leakage pathway, a portion of the ECCS recirculation is assumed to leak into the RWST." ... "Leakage to the RWST is modeled at a rate of 3.8 gpm.

Please clarify whether the back-leakage to the RWST is terminated at 16 hours, or it continues after 16 hours. If it is terminated, justify how the valves used to terminate the release meet Regulatory Position 5.2 of RG 1.183, and state whether the surveillance requirement for determining the operability of these valves requires them to have zero leakage in the configuration that would exist during a DBA LOCA.

RAI ARCB1-LOCA-5 - LOCA

Enclosure IV states, in part, the following in the revised USAR Section 6.5.3.1, "Primary Containment":

> The containment walls, liner plate, penetrations, and isolation valves function to limit the release of radioactive materials, subsequent to postulated accidents, such that the resulting offsite doses are less than the guideline values of 10 CFR 50.67.

The WCGS Bases for TS 3.6.3, "Containment Isolation Valves," states, in part:

> The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, La.
>
> The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

> The DBAs that result in a release of radioactive material within containment are a loss-of-coolant accident (LOCA) and a rod ejection accident (Ref. 1)[3]. In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation.

In the response to ARCB-RAI-38 in Enclosure VII by letter dated January 17, 2017, WCNOC states, in part:

> The LOCA and rod ejection dose consequence analyses, which are the only two analyses that model isolation of containment, assumed the containment was isolated at the beginning of event, with the exception of the mini-purge releases for the first 10 seconds of the LOCA event. This is consistent with the current analyses of record for WCGS.

Enclosure IV of the LAR proposes to remove the time to isolation from the above bases. The NRC staff is unable to verify the assumed time to isolation in the LAR within the response to ARCB-RAI-38 (which seems to conflict with the above statement from WCNOC's Bases for TS 3.6.3).

1. Since this timing is used to limit the releases of radioactive materials, subsequent to postulated accidents, such that the resulting offsite doses are less than the guideline values of 10 CFR 50.67, the NRC staff requests that WCNOC provide the assumed time for the containment to isolate after each DBA and describe how these assumptions are considered in the radiological analyses. It does not appear to be realistic to assume the containment is isolated at the beginning of the event unless the containment is not allowed to be unisolated during operations. If this is the case please state so, and justify this answer.

RAI ARCB1-FHA-1 – Fuel Handling Accident (FHA) - EES System Credit

A markup of the TS Bases for LCO 3.7.13, "Emergency Exhaust System (EES)," provided in the May 4, 2017 supplement states, in part:

> For the fuel handling accident (FHA) the Emergency Exhaust System is credited as the release point, but no credit is taken for filtration of the release.
>
> Total system failure could result in the atmospheric release from the auxiliary building exceeding the guideline limits of 10 CFR 50.67 (Ref. 5)[4] in the event of a LOCA or fuel handling accident.

A note in WCGS TS LCO 3.7.13, "Emergency Exhaust System (EES)," allows the auxiliary building or fuel building boundary to be opened intermittently under administrative controls during times an FHA could occur (for example, "During the movement of irradiated fuel assemblies in the fuel building").

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[3] USAR Section 15.

[4] 10 CFR 50.67, "Accident source term."

The Enclosure IV table entitled "Regulatory Guide 1.194 Comparison" states that the analysis in the LAR conforms with RG 1.194, Regulatory Position 3.2.4.2, which states:

> Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used. Releases from personnel air locks and equipment hatches exposed to the environment, or containment purge releases prior to containment isolation, may need to be treated differently. It may be necessary to consider several cases to ensure that the χ/Q value for the most limiting location is identified.

In WCNOC's response by letter dated May 4, 2017, to the supplemental information request (pages 2 and 3 of the attachment), WCNOC states, in part:

> Thus, the current note in TS 3.7.13, which allows for the fuel building boundary to be opened intermittently under administrative controls, ensures that any fuel building boundary opening will be rapidly closed at the beginning of an FHA. For the proposed changes, the rapid closure will continue to ensure that the fuel building boundary will be operable for the duration of the FHA and therefore the proposed WCGS Technical Specifications are consistent with the analysis discussed in Section 4.3.12 of Enclosure IV to ET 17-0001 [letter dated January 17, 2017].

RG 1.183, Regulatory Position 5.1.4, "Applicability of Prior Licensing Basis," states, in part:

> The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a <u>current finding</u> of compliance with regulations applicable to the amendment.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the AST analysis conforms to Regulatory Position 5.1.4.

Section 50.36 of 10 CFR, requires the TSs to be derived from the analyses and evaluations included in the safety analysis report. Per WCGS TS Bases B 3.7.13, the EES design basis is established by the consequences of the limiting DBAs, which includes an FHA.

The note in TS LCO 3.7.13 does not appear to be derived from the proposed FHA analysis as required by 10 CFR 50.36. The LAR analysis credits the fuel building boundary at the beginning of the accident, but per TS LCO 3.7.13 the fuel building boundary is allowed to be open and not be restored.

Also, the LAR states that it conforms to RG 1.194, Regulatory Position 3.2.4.2, but no other release points other than through the EES are considered. Per the proposed TS Bases for LCO 3.7.13, the proposed FHA analysis credits the EES system for the release point from a postulated FHA. This credit begins at the start of the event. However, if the fuel building boundary is opened (as allowed by the note), the release would not be assured to go through

the EES, but could go through any openings in the fuel building until the fuel building boundary integrity is restored.

WCNOC has proposed to revise several design basis radiological analyses and the NRC staff must make a current finding of reasonable assurance that adequate protection will be maintained with the fuel building envelope boundaries open and allowed to remain open for the duration of the accident (consistent with the TS discussed above). That assurance cannot be based solely on the probability of the accident not occurring since the applicable safety analysis for these TSs is based upon the fundamental assumption that the DBA occurs. The reasonable assurance assessment will include a determination of whether the limits in GDC 19 of Appendix A to 10 CFR Part 50, and 10 CFR 50.67 are met. In order for the NRC staff to make its assessment of reasonable assurance please provide the following information:

1. Submit for the NRC staff's review a revised radiological consequences analysis of a FHA that supports the fuel building and auxiliary building boundary being open under administrative control for the duration of the accident (in addition to any other boundaries allowed to be open) to justify the most severe radiological consequences from an FHA. In addition, provide the inputs, assumptions, methodology technical basis for the analysis and the results of the analysis. Also, please justify the assumptions and inputs used in the analysis.

   or

   Provide a proposed change to the LCO note so that it is consistent with proposed radiological consequence analyses and ensures that the fuel and auxiliary building boundaries are restored consistent with the shortest restoration time evaluated in the licensing basis analyses (including, as applicable, any consideration for obtaining the design basis pressures assumed in these analyses).

RAI ARCB1-FHA-2 - FHA

In the NRC staff's RAI ARCB-RAI-20, discussed in Enclosure VII to the letter dated January 17, 2017, the NRC staff requested data for the current fuel types used at WCNOC that justify a DF of 200 for fuel pressures up to 1500 pounds per square inch gauge (psig) and a detailed justification for using a DF of 200 for pressures up to 1500 psig.

In WCNOC's response to ARCB-RAI-20, the licensee states the following:

1. The current fuel type for WCNOC (17X17 RFA-2) is generically addressed for a DF of 200 at higher rod internal pressures by the approved WCAP-16072-P-A ("Implementation of Zirconium Diboride Burnable Absorber Coatings in CE [Combustion Engineering] Nuclear Power Fuel Assembly Designs" (ADAMS Accession No. ML042510053)).

2. The approval of the WCAP-16072-P-A topical report was based upon evaluations performed in WCAP-7518-L (Legacy Accession No. 9804290400), which WCNOC asserts is not fuel type specific. Therefore, WCNOC asserted that the justification in WCAP-16072-P-A is applicable to all Westinghouse fuel types.

3. The NRC staff had previously approved the use of a DF of 200 for fuel pressures up to 1500 psig in a safety evaluation for Indian Point (ADAMS Accession No. ML050750431).

The NRC is concerned that the information provided in ARCB-RAI-20 does not adequately address the ARCB-RAI-20 request for information. In the cover letter for the NRC staff's safety evaluation for WCAP-16072-P (ADAMS Accession No. ML041270102), it is stated, in part:

> The staff has found that WCAP-16702-P, Revision 00, is acceptable for referencing in licensing applications for CE Nuclear Power designed pressurized water reactors to the extent specified and under the limitations delineated in the report and in the enclosed SE [safety evaluation]. The SE defines the basis for acceptance of the report.
>
> Our acceptance applies only to material provided in the subject TR [topical report]. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

WCNOC is not a CE Nuclear Power designed PWR and, therefore, the staff's acceptance of WCAP-16702-P-A, Revision 00 is not applicable to WCNOC.

Secondly, the NRC staff believes that WCAP-7518-L was not approved by the NRC staff, and the experimental tests were, in part, performed using equipment that simulated the cross-section of a full-scale 14x14 assembly. Therefore, the report results are based upon a fuel design that is not the same as the 17X17 RFA-2 fuel and no basis for its validity for these fuel designs is provided.

Lastly, while the Indian Point safety evaluation review does discuss the DF assumed by Indian Point, and the NRC staff used the Indian Point assumed DF in its analysis, the staff did not provide an explicitly documented review of Indian Point's assumption. No basis for the NRC staff's use of the DF of 200 is provided in the safety evaluation.

1. WCNOC is requested to provide the data for current fuel types used at WCGS 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.

RAI ARCB1-FHA-3 - FHA

RG 1.183, Appendix B, 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 the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel

> rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.

Enclosure IV, Table C, "Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)," states that the analysis conforms to Regulatory Position 1.1 and provides the following comment:

> For the postulated fuel handling accident, one entire assembly and 20% of an adjacent assembly were assumed to be damaged as a result of this event.

RG 1.183, Regulatory Position 5.1.3, "Assignment of Numerical Input Values," states, in part:

> The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the AST analysis conforms to Regulatory Position 5.1.3, which states:

> The numeric values that were chosen as inputs to the analyses required by 10 CFR 50.67 were selected with the objective of determining a conservative postulated dose.
>
> For a range of values, the value that resulted in a conservative postulated dose was used.

RG 1.183, Regulatory Position 5.1.4, "Applicability of Prior Licensing Basis," states, in part:

> The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a <u>current finding</u> of compliance with regulations applicable to the amendment.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the AST analysis conforms to Regulatory Position 5.1.4.

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

> Credit may be taken for accident mitigation features that are classified as safety-related, are <u>required to be operable by technical specifications</u>, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," the licensee states that the WCGS analysis conforms to RG 1.183, Regulatory Position 5.1.2.

Core Alteration is defined, in part, in the WCGS TSs as:

> CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel.

The WCNOC proposed FHA analysis assumes the dropping of a single irradiated fuel assembly, but does not appear to be consistent with Regulatory Position 1.1 because the analysis does not consider dropping other loads such as those allowed to be moved during Core Alterations. Because safety systems credited in the proposed WCGS analyses may not be required to be operable during movement of these loads, per RG 1.183, Regulatory Position 5.1.2, they would not be credited to mitigate this accident. Because the NRC staff must make a current finding of compliance with the regulations and it is unclear whether the most conservative postulated dose has been determined, the NRC staff requests the following:

1. Please provide an evaluation (results, inputs, assumptions, and justifications for these assumptions) performed for dropping of loads allowed over irradiated fuel assemblies (i.e., a new fuel assembly, sources, or reactivity control components) onto irradiated fuel assemblies in the reactor vessel or fuel storage pool and confirm that the resulting onsite and offsite dose results are bounded by the proposed fuel handling accident when crediting only those safety systems required to be operable by the WCGS TSs.

RAI ARCB1-FHA-4 - FHA

RG 1.183, Appendix B, Regulatory Position 1.2, states:

> The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.

Enclosure IV, Table 4.3-15, "Assumptions Used for Fuel Handling Accident Analysis," lists the source term for the FHA. This list includes the following isotopes:

- Krypton (Kr)-85m, Kr-85
- Xenon (Xe)-131m, Xe-133, Xe-133m, Xe-135, Xe-135m
- Iodine (I)-130, I-131, I-132, I-133, I-135

Enclosure IV, Table C, "Conformance with Regulatory Guide 1.183 Appendix B (Fuel Handling Accident)," states that the analysis conforms with Regulatory Position 1.2 and provides the following comment:

> The fission product release from the breached fuel was based on Regulatory Position 3.2 of RG 1.183 and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods was assumed to be instantaneously released. Radionuclides that were considered include xenons, kryptons, halogens, cesiums, and rubidiums.

1. The source term provided in Table 4.3-15 does not appear to conform to Regulatory Position 1.2 because the consideration of cesium and rubidium is not included in the analysis. Please justify the deviation from the RG 1.183 or conform to RG 1.183.

RAI ARCB1-FHA-5 - FHA

On Pages 5, 6, and 7 of the Attachment to the letter dated May 4, 2017, the licensee provided a discussion of an FHA analysis case with the containment open and a release pathway from the containment to the auxiliary building. It states, in part:

> Since containment is not pressurized during a FHA, there may be two scenarios that act as a driving force for activity.
>
> For both scenarios, the containment is assumed to be isolated 2 hours after event initiation based on administrative controls maintained by Wolf Creek [WCGS] to close the containment penetrations following a FHA inside containment.
>
> A conservative auxiliary building volume of 7.0E+04 ft^3 is modeled based on the volume of the rooms located between the containment penetration and the control room.

RIS 2001-019, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests" (ADAMS Accession No. ML011860407), states, in part:

> The NRC staff reviews licensee amendment requests to ensure that the proposed change will maintain an adequate level of protection of public health and safety. The NRC staff accomplishes these reviews by evaluating the information submitted in the amendment request against the current plant design basis as documented in the Final Safety Analysis Report (FSAR), previously issued staff safety evaluation reports, regulatory guidance, other licensee commitments, and staff experience gained in considering similar requests for other plants. The NRC staff bases its finding on the acceptability of an amendment on its assessment of the licensee's analysis, since it is the licensee's analysis that becomes part of the facility's design basis. Licensees should ensure that adequate information, including analysis assumptions, inputs, and methods is presented in the submittal to support a staff assessment. The NRC staff's assessment may include performance of independent analyses to confirm the licensee's conclusions. Licensees should expect an NRC staff effort aimed at resolving critical differences between analysis assumptions, inputs, and methods used by the licensee and those deemed acceptable to the NRC staff.

RIS 2006-04, "Experience with Implementation of Alternative Source Terms," states, in part:

> An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each

individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable.

Additional details are needed to allow the NRC staff to confirm the dose analysis results for the FHA. With the containment penetrations assumed to be open for 2 hours after an FHA, winds or possible "stack effects" could cause flow into the containment and could provide a driving force for radioactivity to move from the containment into the auxiliary building. The flow of air may not be bounded by the assumed flowrate of the heating, ventilation, and air conditioning (HVAC) operating in the auxiliary building or the leakage into the control room. The driving force of the air into the auxiliary building also may cause the auxiliary building pressures around the control room boundary to be higher than those when the inleakage testing was performed.

1. Please provide justification for the assumptions made regarding the flows assumed into the auxiliary building, the dilution volume credited for the auxiliary building, and the unfiltered inleakage into the control room considering the possible environmental conditions due to winds entering the open containment penetrations or "stack effects" in the containment, or revise the assumptions and provide a justification for the new assumptions. Please consider all the different configurations for containment openings allowed by your TSs. Note that RG 1.183 allows mixing in other volumes such as the containment (up to 50 percent of the free volume) and the fuel building on a case-by-case basis, but no guidance exists for mixing in the auxiliary building.

RAI ARCB1-FHA-6 - FHA

By letter dated May 4, 2017, WCNOC provided a response to RAI 3 (concerning the NRC staff's request for a detailed summary of the radiological consequences of an FHA in containment that supports the containment penetrations being open under administrative control and closed during an accident to justify the most severe radiological consequences from an FHA). The licensee's response stated, in part:

> The fuel handling accident (FHA) doses in the alternate source term (AST) license amendment request (LAR) consist of a simplified model of release from the fuel building to the environment through the unit vent. This bounds a similar release from the containment equipment hatch to the environment for a FHA occurring inside containment.
>
> For the case of the containment being open, the release pathway is from containment to the auxiliary building through a limiting composite pathway and from the auxiliary building to the control room via inleakage. The limiting composite pathway considered a conservatively low mixing volume available in both containment and the auxiliary building, the maximum expected penetration flow area available, and the minimum travel distance of all containment penetrations able to be unisolated under administrative controls and the personnel hatch. The containment volume of 1.25E+06 ft^3 is 50% of the minimum free containment volume per RG 1.183, Appendix B, Position 5.5. A

> conservative auxiliary building volume of 7.0E+04 ft^3 is modeled based on the volume of the rooms located between the containment penetration and the control room.

The Enclosure IV table entitled "Regulatory Guide 1.194 Comparison" states that the analysis in the LAR conforms with RG 1.194, Regulatory Position 3.2.4.2, which states:

> Since leakage is more likely to occur at a penetration, analysts must consider the potential impact of building penetrations exposed to the environment within this modeled area. If the penetration release would be more limiting, the diffuse area source model should not be used. Releases from personnel air locks and equipment hatches exposed to the environment, or containment purge releases prior to containment isolation, may need to be treated differently. It may be necessary to consider several cases to ensure that the χ/Q value for the most limiting location is identified.

Based on the above, it appears that the licensee has analyzed a release pathway from the containment to the auxiliary building, but not the other potential release pathways and not the various combinations of allowed open and closed penetrations. The NRC staff needs additional information regarding how the proposed FHA bounds the potential FHA in the reactor containment with the many different containment configurations of penetrations allowed open during Core Alterations and during the movement of irradiated fuel assemblies within containment, including the containment purge, equipment hatch, personnel air lock and other containment penetrations.

1. Please provide a detailed summary of the radiological consequences of an FHA in containment with each penetration allowed to be open and with the various combinations of penetrations allowed to be open to justify the most severe radiological consequences from an FHA. Please show that the dose results for these scenarios meet the limits in GDC 19 of Appendix A to 10 CFR Part 50 and 10 CFR 50.67. In addition, please provide the inputs, assumptions, methodology a technical basis for the analysis, and justify the assumptions used.

RAI ARCB1-FHA-7 - FHA

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

> 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.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," the licensee states that the WCGS analysis conforms to RG 1.183, Regulatory Position 5.1.2.

A markup of the TS Bases for LCO 3.7.13, "Emergency Exhaust System (EES)," provided in Enclosure III to letter dated May 4, 2017, states, in part:

> For the fuel handling accident (FHA) the Emergency Exhaust System is credited as the release point, but no credit is taken for filtration of the release.

In the NRC staff's RAI ARCB-RAI-19, discussed in Enclosure VII to the letter dated January 17, 2017, the NRC staff requested an explanation of the worst case single failure that is assumed for the FHA in the proposed analysis and its justification.

WCNOC's response states:

> The current single failure of the humidity control system resulted in a reduced iodine removal efficiency. The fuel handing accident analysis, documented in Section 4.3.12 of Enclosure IV of this LAR, modeled a direct release of activity to the environment with no credit for the filtration of the EES. Therefore, by conservatively not crediting the EES filtration, the analysis results documented in Section 4.3.12 of Enclosure IV of this LAR are more limiting than if the previous single failure of the humidity control system had been retained and a reduced iodine removal efficiency was credited.
>
> Consistent with the current control room radiological consequences calculation models, as discussed on page 15A-8 of the USAR, a failure of one of the filtration fans is assumed at the start of emergency mode of operation and a larger unfiltered inflow to the control room is assumed since only half of the makeup flow to the control room can pass through a filter. After a defined time of 90 minutes, operator action isolates the failed train and reduces the unfiltered inflow to the control room.

In the NRC staff's RAI ARCB-RAI-30, discussed in Enclosure VII to the letter dated January 17, 2017, the NRC staff requested the worst case single failure for each accident.

WCNOC's response states:

> Consistent with the current control room radiological consequences calculation models, as discussed on page 15A-8 of the USAR, a failure of one of the filtration fans is assumed at the start of emergency mode of operation and a larger unfiltered inflow to the control room is assumed since only half of the makeup flow to the control room can pass through a filter. After a defined time of 90 minutes, operator action isolates the failed train and reduces the unfiltered inflow to the control room. The following accidents modeled switchover to the CREVS [Control Room Emergency Ventilation System] and included the failure of one of the filtration fans: [the list included the Fuel Handling Accident]

The NRC staff does not believe that the response for ARCB-RAI-19 justifies the worst case single failure assumed for the FHA. The justification for not retaining the single failure of the humidity control system relies on the modeling assumption that the EES filtration is not credited.

1. Is the EES not being credited the worst case single failure for the FHA or is it the failure of one of the filtration fans at the start of the emergency mode of operation?

2. If failure of the EES is not the worst case single failure, please identify the single failure (failure of the humidity control system or failure of one of the filtration fans at the start of the emergency mode of operation) that results in the maximum postulated doses? If the worst case single failure is the failure of the humidity control system, then provide an FHA analysis (inputs, assumptions and results) for this scenario.

RAI ARCB1-SGTR-1 – Steam Generator Tube Rupture (SGTR)

RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," Regulatory Position 5.6, provides guidance for the modeling of the transport of radioactivity after a SGTR. Table E, "Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)" of Enclosure IV, in the LAR, states that the SGTR analysis conforms to RG 1.183, Appendix F, Regulatory Position 5.6 [points to Regulatory Positions 5.5 and 5.6 of Appendix E], which states:

> The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the SGTR. In addition, flashing of break flow in the ruptured steam generator with a time dependent flashing fraction was considered and all activity in the flashed break flow was released to the environment with no mitigation, dilution, or credit for scrubbing.

Enclosure IV, Section 4.3.8.2.2, "Release Model," states, in part:

> Iodine and alkali metal activity contained in the portion of the break flow that flashed to steam upon entering the ruptured steam generator [SG] is released directly to the atmosphere as long as steam releases from the ruptured SG continue. An iodine partition coefficient in the SGs of 100 (Ci [Curies] iodine/gm [gram] water) / (Ci iodine/gm steam) is applied to releases resulting from steaming of the secondary side fluid.

1. Please confirm whether the partition coefficient of 100 is applied only to the radioactivity in the bulk water assumed to become vapor by steaming and not to any primary-to-secondary flow that flashed to steam.

RAI ARCB1-SGTR-2 - SGTR

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

> Assumptions regarding the <u>occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences</u>.

In Table A, "Conformance with Regulatory Guide 1.183 Main Section," of Enclosure IV, the licensee states that the WCGS analysis conforms to RG 1.183, Regulatory Position 5.1.2, which states, in part:

> Assumptions regarding the occurrence and timing of a loss of offsite power were also selected with the objective of maximizing the postulated radiological consequences.

RG 1.183, Appendix F, Regulatory Position 5.4 provides guidance for the modeling of the transport of radioactivity after a SGTR and states:

> The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

Enclosure IV, Table E, "Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)," states that the SGTR analysis conforms to RG 1.183, Appendix F, Regulatory Position 5.4, but also states:

> A loss of offsite power was assumed coincident with reactor trip.

The reactor trip is assumed to occur 52 seconds after the SGTR. Assuming the loss of offsite power coincident with the reactor trip would allow credit for the condenser before the loss of offsite power. The assumption does not seem to be selected with the objective of maximizing the postulated radiological consequences, and the release of fission products from the secondary would not be coincident with the loss of power.

1. Please justify how the SGTR conforms to these regulatory positions or revise the analysis to be consistent with them.

RAI ARCB1-SGTR-3 - SGTR

Enclosure IV, Section 4.3.8.2.2, "Release Model," states, in part:

> The entire 1 gpm primary-to-secondary accident-induced leakage allowed by the TS is assumed to be leaking into the intact [unaffected] SGs with a density based on cooled liquid, which otherwise is negligible compared to the flow through the ruptured tube.

The "Analysis" column in Table E "Conformance with Regulatory Guide 1.183 Appendix F (PWR Steam Generator Tube Rupture Accident)," of Enclosure IV states the analysis conforms to Regulatory Position 5.1

RG 1.183, Appendix F, Regulatory Position 5.1, states:

> The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

In the LAR, the WCGS analysis proposes to change the assumption that the entire 1 gpm primary-to-secondary leakage is goes through the affected generator to the assumption that it goes through the unaffected generators. The LAR states that this leakage modeling was used since the leakage is negligible compared to the flow through the ruptured tube. While this is true, the assumption may not be conservative.

RG 1.183, Appendix F, Regulatory Position 5.1, states that the leakage should be apportioned between the affected and unaffected steam generators in a manner which maximizes the calculated dose.

1. Please justify how apportioning the primary-to-secondary leak rate to only the intact steam generators maximizes the accident dose, or apportion the primary-to-secondary leakage to the steam generators that maximize the accident dose and justify any proposed changes from the current licensing bases.

RAI ARCB1-SGTR-4 - SGTR

Enclosure IV, Section 4.3.8.2.2, "Release Model," for the SGTR accident states, in part:

> The entire 1 gpm primary-to-secondary accident-induced leakage allowed by the TS is assumed to be leaking into the intact SGs with a density based on cooled liquid, which otherwise is negligible compared to the flow through the ruptured tube.

Consistent with the Standard TSs, the "Applicable Safety Analysis" section of the bases for TS 3.4.13, "RCS Operational LEAKAGE" currently states, in part:

> The safety analyses for events resulting in steam discharge to the atmosphere assume that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions. [Note the LAR proposes to remove the words "or increases to one gallon per minute."]
>
> The safety analysis for the SLB accident assumes the entire 1 gpm primary to secondary LEAKAGE is through the affected generator as an initial condition.

The Standard TSs, "Applicable Safety Analysis" section of the bases for TS 3.4.16, "RCS Specific Activity" states:

> The SGTR safety analysis (Ref. 2)[5] assumes the specific activity of the reactor coolant at the LCO [limiting condition for operation] limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm.

The LAR proposes to change the existing bases as follows:

> The safety analyses (Refs. 3 and 4)[6] assume the specific activity of the reactor coolant is at or more conservative than the LCO limits, and ~~an existing~~ reactor

[5] FSAR Section [15.6.3].
[6] USAR, Sections 15.1.5 and 15.6.3.

coolant steam generator (SG) tube leakage rate of 1 gpm exists or results from accident induced conditions.

1. Based upon the changes in Enclosure IV of the LAR stated above, these accidents will only consider the accident induced leakage rather than also considering a pre-existing leakage limit of 1 gpm. If operation with a 1 gpm primary-to-secondary leakage is allowed per the WCGS TSs, justify why the 1 gpm primary-to-secondary leakage is not assumed in the accident analyses.

RAI ARCB1-SGTR-5 - SGTR, MSLB, and other accidents that assume DEX 133

Enclosure IV of the LAR states that the definition of DEX 133 in the WCGS TSs is proposed to be revised to reflect the RCS dose equivalent noble gas curie content for all dose analyses modeling initial RCS activity. Enclosure IV also proposed revising several DBAs including the MSLB and the SGTR, which use the value of the Dose Equivalent Xe-133 specified in TS 3.4.16, "RCS Specific Activity."

The proposed TS Bases for TS 3.4.16, states, in part:

> The maximum dose that an individual at the exclusion area boundary can receive for any 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1)[7]. Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.
>
> The analysis for the SLB [steamline break] and SGTR accidents establish the acceptance limits for RCS specific activity.
>
> The analyses consider two cases of reactor coolant specific activity. ... In both cases, the noble gas specific activity is assumed to be equal to or greater than 500 µCi/gm [micro-Curies per gram] DOSE EQUIVALENT XE-133.

Section 50.67 of 10 CFR requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs.

Section 50.36 of 10 CFR requires the TSs to be derived from the analyses and evaluation included in the safety analysis report, which include the SLB and SGTR.

Per WCGS, the proposed TS 3.4.16, "RCS Specific Activity," when the DEX 133 is not within limit (500 micro-Curies per gram) the RCS DE I-131 is to be restored to within the limit within 48 hours.

WCNOC has voluntarily proposed to revise the design basis radiological analysis under 10 CFR 50.67. In the proposed design bases analysis the condition allowed in the WCGS TSs, which allows continued operation with the DEX 133 at levels higher than 500 micro-Curies per gram, is an unanalyzed condition where compliance with 10 CFR 50.67 is unknown. Note that

[7] 10 CFR 50.67, "Accident source term."

the current and proposed TS 3.4.16 Bases states that these analyses assumed values equal to or greater than 500 micro-Curies per gram DEX 133.

The NRC staff must make a current finding of reasonable assurance that adequate protection will be maintained during the conditions allowed by the TSs. That assurance cannot be based solely on the probability of the accident not occurring since the applicable safety analysis for these TSs is based upon the fundamental assumption that the DBA occurs. Please note that the NRC staff sent a letter dated April 27, 2016, to the Technical Specification Task Force regarding an NRC staff issue with the 48 hour completion time for DE XE-133 (ADAMS Accession No. ML16113A402). Therefore, WCNOC needs to provide an analysis consistent with the proposed TSs that shows that the limits in GDC 19 of 10 CFR 50 Appendix A and 10 CFR 50.67 are met. In order for the staff to make its assessment of reasonable assurance please provide the following information:

1. For every accident that assumes the RCS activity is based upon the value of the DEX 133 specified in TS 3.4.16, "RCS Specific Activity" please submit for the NRC staff's review a revised radiological consequences analyses that assumes the DEX 133, allowed by the proposed TSs (values equal to or greater than 500 micro-Ci/gm) at the start of the event and show that the dose results meet the limits in GDC 19 of Appendix A to 10 CFR Part 50 and 10 CFR 50.67. Note this case would be consistent with the proposed and current TS Bases, which states that: In both analyzed cases for the noble gas specific activity is assumed to be equal to or greater than 500 μCi/gm DOSE EQUIVALENT XE-133. In addition, provide the inputs, assumptions, methodology technical basis for the analysis and the results of the analysis. Also, please justify the assumptions and inputs used in the analysis.

   or

2. Please provide a proposed change to TS 3.4.16 that is consistent with the analyses proposed in the LAR. Note that an example of what has been found acceptable to the NRC staff can be seen with the treatment of Dose Equivalent I-131 in TS 3.4.16. In this treatment, when values of RCS activities are greater than those analyzed in the DBA analyses (60 micro-Curies/gm) the required action is to begin immediate shutdown of the reactor within 6 hours (See is Condition C of TS 3.4.16).

RAI ARCB1-SGTR-6 - SGTR

Page 71 of Enclosure VII to the letter dated January 17, 2017, states, in part:

> Note that Enclosure IV, Section 4.3.2.1 describes that isolation based solely on the control room radiation monitor does not fully switch the HVAC systems such that the control room unfiltered inleakage continues to be associated with the normal mode χ/Qs.

Enclosure IV, Footnote 5 of Table 4.3-11, "Assumptions Used for the SGTR Dose Analysis," states:

> The control room unfiltered inleakage continues to be associated with the normal mode χ/Q of 2.55E-02 sec/m3 [seconds per cubic meter] until completion of control room isolation from a safety injection signal at 10 minutes. [Note: The

> 2-8 hour control room atmospheric dispersion factor given is 1.04E-3 seconds/cubic meter. This corresponds to the atmospheric dispersion factor for the emergency control room intake given in Table 4.1.2-3(a)].

The NRC staff could not locate information regarding the source of the unfiltered inleakage into the control room. Unfiltered inleakage can enter into the control room through ductwork and openings in the control room boundary, but the LAR appears to only model inleakage through the intake ducts.

1. Please justify the use of the emergency and normal intakes atmospheric dispersion factors and why they are limiting for unfiltered inleakage into the WCGS control room (which could also come into the control room from locations other than the intake ducts).

RAI ARCB1-LRA-1 - Locked Rotor Accident (LRA)

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

> Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," the licensee states that the WCGS analysis conforms to RG 1.183, Regulatory Position 5.1.2, which states, in part:

> Assumptions regarding the occurrence and timing of a loss of offsite power were also selected with the objective of maximizing the postulated radiological consequences.

RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," Regulatory Position 5.4, provides guidance for the modeling of the transport of radioactivity after a Locked Rotor accident and states:

> The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

Enclosure IV, Table F, "Conformance with Regulatory Guide 1.183 Appendix G (PWR Locked Rotor Accident)," states that the LRA analysis conforms to RG 1.183, Appendix G, Regulatory Position 5.4, but then states:

> A loss of offsite power was assumed.

1. Please clarify when the loss of offsite power is assumed and justify how this conforms to Regulatory Positions 5.1.2 and 5.4 discussed above.

RAI ARCB1-MSLB-1 - Main Steam Line Break (MSLB)

Page 15.1-14 of the USAR in Enclosure IV states, in part:

> The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:
>
> Assuming a stuck RCCA [rod cluster control assembly] with or without offsite power, and assuming a single failure in the engineered safety features system, the core cooling capability is maintained.

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

> 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.

In Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," the licensee states that the WCGS analysis conforms to RG 1.183, Regulatory Position 5.1.2.

1. Please state if the loss of offsite power was assumed to maximize the postulated MSLB radiological consequences. If a methodology other than that in RG 1.183 is used, please provide details about the methodology and justify its use, and why it is conservative.

RAI ARCB1-MSLB-2 - MSLB

Enclosure IV, Section 4.3.3.2.3, "Control Room" states, in part:

> In the event of an MSLB, the low steamline pressure SI setpoint <u>will be reached almost immediately</u> following the break.

USAR Section 15.1.5, "Steam System Piping Failure" states, in part:

> During startup or shutdown evolutions, when the operator manually blocks the safety injection on low pressurizer pressure or low steamline pressure and steamline isolation on low steamline pressure when pressurizer pressure is less than P-11 setpoint (i.e., 1970 psig), the steamline pressure-negative rate-high signal is automatically enabled to provide steamline isolation. For inside containment breaks, steamline isolation may also be provided by the containment pressure High-2 signal and safety injection would be actuated by the containment pressure High-1 signal. <u>For a steamline break occurring outside containment, an automatic actuation signal for safety injection would not be available.</u>

> Note that the conclusion that the hot-zero power case is the limiting case is based on certain specific protection system performance characteristics credited for "at power" steamline break analyses.

RG 1.183, Regulatory Position 5.1.3, "Assignment of Numeric Input Values," states, in part:

> The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the AST analysis conforms to Regulatory Position 5.1.3, which states:

> The numeric values that were chosen as inputs to the analyses required by 10 CFR 50.67 were selected with the objective of determining a conservative postulated dose.
>
> For a range of values, the value that resulted in a conservative postulated dose was used.

Per the USAR, during shutdown and startup evolutions the automatic actuation signal for SI would not be available, but the proposed MSLB analysis assumes the SI setpoint would be reached almost immediately. This assumption may result in non-conservative radiological doses, which would be inconsistent with Regulatory Position 5.1.3 (contrary to WCGS stated conformance to Regulatory Position 5.1.3).

1. Please state the assumed time for the SI setpoint to be reached and state the reference analysis used to determine this value. Justify how assuming this time results in the worst case radiological consequences and why the SI signal is credited when the USAR says there are conditions when it would not be available.

RAI ARCB1-WT-1 - Gaseous Waste Tank Failure and Recycle Holdup Tank

RG 1.183, Regulatory Position 4.1.1, states, in part:

> The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the proposed analysis conforms with Regulatory Position 4.1.1. Tables 4.3-2a "Tank Activities – AST" and 4.3-2b, "Tank Activities – CLB" of Enclosure IV provide activities for the Waste Gas Tank and the Recycle Holdup Tank for the LAR and these activities for the current licensing basis. In the current licensing basis, Kr-83m, Kr-89 and Xe-137 are included in the determination of the dose, but are not included in the LAR's source term.

1. Please justify the exclusion of these radionuclides in the radiological consequences calculated for the gaseous waste and recycle holdup tank failure analysis.

RAI ARCB1-WT-2 - Waste Gas Decay Tank Failure

Enclosure IV, Table 4.3-13, "Assumptions Used for Waste Gas Decay Tank Analysis," states that the control room and TSC atmospheric dispersion factors assumed are 7.17E-4 and 1.97E-4 sec/m^3, respectively. Per Table 4.1.2-3(b), "Calculated x/Q[8] (s/m^3 [seconds per cubic meter]) Values for the Normal Control Room Air Intake," these atmospheric dispersion factors correlate to releases from the Radwaste Building to the Normal Control Room intake from 0 to 2 hours.

USAR Section 11.3.3.2, "Release Points," states:

> Release Points from the gaseous waste processing systems are shown on Figure 11.3-2.

Inspection of Figure 11.3-2, "Potential Gaseous Release" shows that there are other possible structures that contain the gaseous waste processing system and other potential release points than the Radwaste Building. However, these potential release points do not seem to be considered in the limiting tank failure analysis because the more limiting atmospheric dispersion factors for these release points were not used in the analysis.

1. Please justify why the waste gaseous decay tank failure analysis only considers a release from the Radwaste Building and not the other potential release points detailed in USAR Figure 11.3-2.

RAI ARCB1-WT-3 - Waste Gas Decay Tank Failure and Liquid Waste Tank Failure

Enclosure IV, Section 4.3.10.2.1, "Source Term," states, in part:

> The iodine is assumed to be 100% elemental; however, the chemical species of iodine has no impact on the calculation since no removal processes are modeled <u>and the control room filters have the same efficiencies for all forms of iodine.</u>

Enclosure IV, Section 4.3.10.2.3, "Control Room," states:

> The control room is not credited to isolate following a tank failure; therefore, the control room ventilation remains in normal operation mode. This modeling is conservative since activity <u>reduction due to filtration of inflow and filtered recirculation is not credited</u>.

The above statements seem to conflict and do not consider the impact of this assumption on the partition factor assumed. If the <u>control room filters are not credited</u> (as stated in Section 4.3.10.2.3), the statement in Section 4.3.10.2.1 that the chemical species of iodine has no impact on the calculation <u>since the control room filters have the same efficiencies</u> for all forms of iodine, needs to be clarified. Like noble gases, organic iodine is not likely to be retained in the water; therefore the partition factor for elemental iodine is expected to be higher than that for organic iodine and, therefore, the speciation of iodine assumed can impact the doses calculated.

---

[8] Atmospheric dispersion factors

1. Please revise the justification for using 100 percent elemental iodine in these analyses to clarify whether control room filters were credited or not and why the use of 100 percent elemental iodine is conservative and justified.

RAI ARCB1-WT-4 - Liquid Waste Tank Failure (LWTF)

Enclosure IV, Section 4.2, "Accident Source Terms," states, in part:

> The hypothetical liquid waste tank inventory is based on a series of hand calculations and is intended to bound the inventory of several smaller waste tanks (such as the recycle holdup tank, waste holdup tank, and floor drain tank). The results presented in Table 4.2-5 are the maximum values calculated for each radionuclide.

1. Please fully describe the "hand calculations" in enough detail so that the NRC staff can verify the results of the calculation.

RAI ARCB1-WT-5 - Waste Gas Decay Tank

RIS 2006-04, "Experience with Implementation of Alternative Source Terms," states, in part:

> An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable.

In Section 8, "NRC Regulatory Issue Summary 2006-04 Comparison," of Enclosure IV, WCNOC states the submittal conforms to this RIS position.

In the NRC staff's RAI ARCB-RAI-31, discussed in Enclosure VII to the letter dated January 17, 2017, the NRC staff requested an explanation of proposed changes to the licensing basis for the waste gas decay tank failure. WCNOC's response states, in part:

> As noted in the above discussion, the methods and assumptions used in defining the activity inventories in the failed gas decay tank are consistent with the assumptions stated in Regulatory Guide 1.24 Section C.1, Regulatory Position; additionally, conservatism has been added by considering all of the activity that is generated during the previous operating cycle, plus the activity that is removed during the shutdown is contained in the failed gas decay tank. The previous assumption of a 40-year accumulation of krypton-85 activity <u>is overly conservative and inconsistent with the design and operation of the waste gas system</u>.

> In addition to the noble gases, radioactive iodine is considered to partition in the VCT [volume control tank] vapor space (a partition factor = 100) and to be carried over into the gas decay tank during the degassing operations. [1 percent of the iodine is released]
>
> Two separate scenarios for the accidental release of activity from liquid waste tanks are analyzed; that is, 1) the failure of a recycle holdup tank from which the release of 100% of the noble gas nuclides and 10% of the iodine activity in the tank is assumed, and 2) the failure of a hypothetical liquid waste tank from which 100% of the iodine activity is released.

Enclosure IV provided two tables, Table 4.3-13 and Table 4.3-14, that state that the iodine chemical form modeled in the Waste Gas Decay Tank Failure analysis and the Liquid Waste Tank Failure analysis assumes that all the iodine in these tanks is assumed to be 100 percent elemental iodine and that this is a change from the current licensing basis.

1. Please justify why the previous assumption of a 40-year accumulation of krypton-85 activity is overly conservative and how it is inconsistent with the design and operation of the waste gas system. Is the tank restricted to only holding 2 cycles of activity?

2. Several different release assumptions for the iodine activity are assumed above. Please justify the partition factor of 1 percent and provide enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. Some factors that should be considered are the pH of the solution, the amount of radioactivity in the solution, and the form of iodine assumed in the liquid. If a change in the assumed form of iodine is made to include organic iodine, please also justify the assumption of a 10 percent release.

RAI ARCB1-CREA-1 - Control Rod Ejection Accident (CREA)

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

> 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.

In Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," the licensee states that the WCGS analysis conforms to RG 1.183, Regulatory Position 5.1.2.

USAR Section 15.4.8.3.1.1, "Physical Model" states, in part:

> Thus, in the case of coincident loss of offsite power, activity is released to the atmosphere from a steam dump through the relief valves. [Note that these words are proposed to be removed as shown in the LAR in Enclosure IV].

1. Please state whether the loss of offsite power was assumed to maximize the postulated Control Rod Ejection Accident radiological consequences. If a methodology other than

that in RG 1.183 is used, please provide details about the methodology and justify the proposed change from the current methodology, and why it is conservative.

RAI ARCB1-CREA-2 - Control Rod Ejection Accident (CREA)

RG 1.183, Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," Regulatory Position 7.4, provides guidance for the modeling of the transport of radioactivity after a CREA. (Note: This regulatory position points to Regulatory Positions 5.5 and 5.6 of Appendix E). Table G, "Conformance with Regulatory Guide 1.183 Appendix H (PWR Rod Ejection Accident)," of Enclosure IV, states that the CREA analysis conforms to RG 1.183, Appendix H, Regulatory Position 7.4, which states:

> The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E for iodine and particulates was considered as appropriate in the rod ejection event.

Enclosure IV, Section 4.3.6.2.2, "Release Model" states, in part:

> An iodine partition coefficient of 100 (Ci iodine/gm [gram] water)/(Ci iodine/gm steam) is applied to releases from the SGs. The release of alkali metals from the secondary side is limited by applying the plant-specific moisture carryover factor of 0.25% to the steam releases. All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

1. Please state what portions of Regulatory Positions 5.5 and 5.6 of Appendix E were considered not appropriate and which were considered for use in the WCNOC CREA analysis. Please justify why they are appropriate. If they are not appropriate, justify the method used.

2. Please confirm whether the partition coefficient of 100 is applied only to the radioactivity in the bulk water assumed to become vapor by steaming and not to any primary-to-secondary leakage that may immediately flash and rise through the bulk water.

RAI ARCB1-CONTROL ROOM-1 - All DBAs

RG 1.183, Regulatory Position 4.2, "Control Room Dose Consequences," provides the guidance for determining the TEDE for persons located in the control room. Regulatory Position 4.2.1, states, in part:

> The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel....
>
> - Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.

In Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," WCNOC states that the analysis conforms to RG 1.183 Regulatory Position 4.2.1 and provided the following comments:

> The TEDE analysis considered all significant sources of radiation that would cause exposure to Control Room personnel. For WCGS, the limiting Control Room dose included:
>
> - Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
> - Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from the Control Building,
> - Radiation shine from the external radioactive plume released from the facility,
> - Radiation shine from radioactive material in the reactor containment,
> - Radiation shine from radioactive material in Control Room recirculation filters and radioactive material in the Control Building.

It appears that the determination of the amount of radiation shine from radioactive material in the control building and its addition to the control room dose consequence may have been missed for some of the proposed DBA analyses.

1. Please explain the methodology used to determine of the amount of radiation shine from radioactive material in the control building to the personnel in the control room and provide the amount of exposure that is added to the control room dose for the following analyses:

   - Fuel Handling Accident (FHA)
   - Main Steam Line Break (MSLB) Accident
   - Steam Generator Tube Rupture (SGTR) Accident
   - Locked Rotor Accident (LRA)
   - Control Rod Ejection Accident (CREA)
   - Letdown Line Break Accident
   - Waste Gas Decay Tank Failure (WGDT)
   - Liquid Waste Tank Failure (LWT)
   - Loss of Non-Emergency AC Power (LOAC)

RAI ARCB1-CONTROL ROOM-2 - Accidents that credit isolation of the control room using the R-23 detector

RG 1.183, Regulatory Position 5.1.3, "Assignment of Numeric Input Values," states, in part:

> The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a

> conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the AST analysis conforms to Regulatory Position 5.1.3, which states:

> The numeric values that were chosen as inputs to the analyses required by 10 CFR 50.67 were selected with the objective of determining a conservative postulated dose.
>
> For a range of values, the value that resulted in a conservative postulated dose was used.

In WCNOC's response to ARCB-RAI-13 discussed in Enclosure VII to the letter dated January 17, 2017, WCNOC assumed the RCS mass of 3.99E+5 pounds mass to calculate the activity concentration at the R-23 detector for several DBAs. This value corresponds to the minimum RCS mass. Enclosure IV also provides a maximum value of 8.42E+5 pounds mass.

1. Please justify the use of the minimum RCS mass in these calculations since the use of the maximum value could result in a more conservative postulated dose.

RAI ARCB1-CONTROL ROOM-3 - Control Room

Paragraph 50.67(b)(2) of 10 CFR requires that the licensee's analysis demonstrates with reasonable assurance that adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

In the LAR, WCNOC has provided the resultant radiation dose associated with occupancy of the control room. However, the LAR appears to be missing a discussion and/or calculation that accounts for the control room personnel radiation exposure received, for the duration of the accident, upon ingress/egress from the site boundary to the control room. In order to meet 10 CFR 50.67 and 10 CFR 50 Appendix A GDC 19, the radiation dose for accessing the control room must be evaluated from the site boundary to the control room for both ingress and egress for the duration of the accident.

1. Please provide an analysis of the radiation dose received from ingress and egress to the control room in enough detail that will enable the NRC staff to be able to perform an independent calculation.

RAI ARCB1-CONTROL ROOM-4 - Control Room and TSC

Chapter 3 of the USAR, Revision 30, Appendix 3A (ADAMS Accession No. ML17151A997), discusses the extent to which WCGS conforms to NRC published regulatory guides. Exceptions to the guides are identified, and justification is presented or referenced. For RG 1.197, no exception to RG 1.197's Regulatory Position 2.5, "Inleakage Test Acceptance Criteria," is noted. This position states, in part:

> Any analysis to demonstrate that a facility meets GDC-19 should include a value for inleakage that is due to ingress to and egress from the CRE [control room envelope]. This value is combined with the baseline test value for inleakage in the analyses. When integrity tests are performed to determine the CRE's integrity characteristics, the acceptance criterion for the test should be the licensing basis amount less the amount designated for ingress and egress. The staff considers 10 cfm [cubic feet per minute] as a reasonable estimate for ingress and egress for control rooms without vestibules.

Furthermore in the report entitled, "WCGS Generating Station Annual Safety Evaluation Report" (ADAMS Accession No. ML010740157), Safety Evaluation Number 59 1999-0148, Revision 0, "USAR Change to Radiological Consequences of a LOCA," states, in part, that "10 cfm control room direct unfiltered inleakage to account for possible increase in air exchange due to ingress or egress," was considered in the LOCA dose analysis.

1. Please clarify if the 10 cfm unfiltered inleakage for ingress and egress from the control room is considered in all the revised radiological analyses incorporating the alternative source term.

2. If not, please either include the 10 cfm unfiltered inleakage or provide a detailed justification why it is appropriate to consider the doors to the control room closed for the duration of the accident and how this would be accomplished considering the need for access to the control room during the accident.

RAI ARCB1-CONTROL ROOM-5 - Control Room and TSC

Enclosure IV, Table 4.2-1, "Core Inventory used in the AST Analysis," provides halogens including both bromine and iodine nuclides that were used in the AST analysis. Table A, "Conformance with Regulatory Guide 1.183 Main Sections," of Enclosure IV states:

> For the LOCA event, the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases in Table 2 were utilized [which provides the release fractions into containment for all halogens].

However, Table 4.3-12, "Assumptions Used for LOCA Analysis," only provides the fuel release fractions and timing for "iodine" rather than halogens.

1. Please provide the revised Table 4.3-12 to make it consistent with the statements made in Table A or provide the release fractions and timing for all halogens (not just iodine).

RAI ARCB1-CONTROL ROOM-6 - Control Room

Per the WCGS TS Bases B 3.7.10, "Control Room Emergency Ventilation System (CREVS)," the CREVS provides airborne radiological protection for the control room envelope occupants, as demonstrated by the occupant dose analyses for the most limiting DBA fission product release presented in the USAR, Chapter 15, Appendix 15A.

A note in WCGS Technical Specification 3.7.10, "CREVS," allows the control room envelope and control building envelope boundaries to be opened intermittently under administrative controls.

The proposed TS Bases, B3.7.10, "CREVS," states, in part:

> In order for the CREVS trains to be considered OPERABLE, the CRE and CBE [control building envelope] boundaries must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBA's, and that CRE occupants are protected from hazardous chemicals and smoke.

RG 1.183, Regulatory Position 5.1.4, "Applicability of Prior Licensing Basis" states in part:

> The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the AST analysis conforms to Regulatory Position 5.1.4.

Enclosure IV, Section 4.3.2.1, "Control Room Model," states, in part:

> At the start of all the events considered, the control room ventilation system is in normal mode. In this mode, unfiltered air from the environment enters the control building and control room. Receipt of a safety injection (SI) actuation signal or a high radiation signal from the control room air intake monitors will isolate the control room and initiate the emergency mode of operation, including a delay.
>
> After [the] emergency mode is initiated, outside air is brought into the control building through safety grade filters. Makeup air is brought into the control room via both trains of the control room filtration system, which draws in air from the control building. Unfiltered air also leaks into the control building and control room via assumed inleakage rates.

For those accidents that do not credit switching to emergency mode, the release pathway into the control room is through the normal ventilation and do not consider pathways into the control room through other locations on the control building or control room boundary.

Section 50.36 of 10 CFR requires the TSs to be derived from the analyses and evaluation included in the safety analysis report. However, the technical analysis section of the LAR does

not include consideration for a scenario where the DBAs occur while the control room and control building envelope boundaries are open and are allowed to remain open (for an indefinite length of time, in addition to its unlimited use), as allowed by the note in Technical Specification 3.7.10.

WCNOC has proposed to revise several design basis radiological analyses and the NRC staff must make a current finding of reasonable assurance that adequate protection will be maintained with the control room and control building envelope boundaries open and allowed to remain open for the duration of the accident (consistent with the TS discussed above). That assurance cannot be based solely on the probability of the accident not occurring since the applicable safety analysis for these TSs is based upon the fundamental assumption that the DBA occurs. The reasonable assurance assessment will include a determination of whether the limits in GDC 19 of Appendix A to 10 CFR Part 50 and 10 CFR 50.67 are met. In order for the staff to make its assessment of reasonable assurance please provide the following information:

1. Submit for the NRC staff's review revised radiological consequences analyses for the design-basis accidents that model the control room. The analyses need to consider a scenario where the design-basis accidents occur while the control room and control building envelope boundaries (in addition to any other boundaries allowed to be open) are open for the duration of the accident and has dose results that meets the limits in GDC 19 of Appendix A to 10 CFR Part 50 and 10 CFR 50.67. In addition, provide the inputs, assumptions, methodology, and the results of the analysis. Also, please justify the assumptions and inputs in used in the analysis.

   or

2. Please provide a proposed change to the LCO note so that it is consistent with proposed radiological consequence analyses and ensures that the control room and control room boundaries are restored consistent with the shortest restoration time evaluated in the licensing basis analyses (including, as applicable, any consideration for obtaining the design basis pressures assumed in these analyses).

RAI ARCB1-GENERAL-1 - Dose Conversion Factors

RG 1.183, Regulatory Position 4.1.4, states:

> The DDE [deep dose equivalent] should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21)[9], provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.

[9] K.F. Eckerman and J.C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states:

> EDE Conversion factors for isotopes were taken from Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil."

Enclosure IV, Table 4.3-3a, "Dose Conversion Factors [DCF] – AST," states that the EDE DCF for Cs-137 is 2.88E-14 Sv-m3/Bq-sec [Sievert-cubic meter per Becquerel-second]. However, the NRC staff's review of the column headed "effective" yield doses in Table III.1 of Federal Guidance Report 12 shows that the DCF for Cs-137 is 7.74E-18 Sv-m3/Bq-sec.

1. Please provide the revised DCF value for Cs-137 in Table 4.3-3a to make it consistent with the statements made in Table A (restated above) or justify the DCF value for Cs-137 and revise the statements in the LAR to make them consistent with the use of a DCF that is different from those in FGR 12.

RAI ARCB1-GENERAL-2 - Several Accidents

RG 1.183, Regulatory Position 5.1.3, "Assignment of Numeric Input Values," states, in part:

> The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," states that the AST analysis conforms to Regulatory Position 5.1.3, which states:

> The numeric values that were chosen as inputs to the analyses required by 10 CFR 50.67 were selected with the objective of determining a conservative postulated dose.
>
> For a range of values, the value that resulted in a conservative postulated dose was used.

RIS 2001-019, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," dated October 18, 2001, states, in part:

> Analysis inputs should be the most restrictive values of plant parameters selected from the range of design values possible during the specific event so that the postulated consequences of the event are maximized. It is generally inappropriate to use values characterized as "best estimates." Licensee commitments to particular regulatory guides and standard review plan sections may establish the value of certain parameters and should continue to be used where applicable. Other considerations follow:

a. The range of values applicable during an accident may vary from accident to accident, and will likely differ from the range that applies during normal operations. For example, a loss-of-offsite-power assumption may affect ventilation system flow rates.

b. It may be necessary to use different parameter values in different portions of the analyses or to perform a sensitivity analysis to determine the limiting value. In some situations the minimum and maximum value of the range may be applicable in a single analysis. For example, the minimum containment spray flow rate is used in determining the spray removal coefficients, but the maximum flow rate may be appropriate in determining the minimum sump pH.

c. If a plant parameter is associated with a technical specification limiting condition for operation (LCO), the value specified in the technical specification should be used. If the LCO specifies a range, or a value with a tolerance band, the most restrictive value should be used. The technical specifications may also specify numeric values in surveillance requirements or action statements; for example, acceptable emergency core cooling system leakage or transient reactor coolant system (RCS) iodine concentration. These should be used where appropriate.

In WCNOC's response to ARCB-RAI-13 (c and d) discussed in Enclosure VII to the letter dated January 17, 2017, the licensee stated:

> A sensitivity analysis was not performed for the normal [control room HVAC] flow rates. However, the actual normal makeup flow rates were measured and compared to the values modeled in the dose analyses. The normal makeup flow rates modeled in the dose analyses are greater than the measured plant flow rates for the control building and control room by more than 10%. [For the fuel handling accident, loss of alternating current power, and letdown line break accidents and tank ruptures].

USAR Section 9.4.1.4, "Tests and Inspections" states, in part:

> During technical specification surveillance testing (TS 5.5.11a., b., f.) of the Pressurization System filter absorber unit, a Pressurization System [filtration] flow rate of 2200 cfm ±10% is verified.

Likewise, TS 5.5.11a, b and f also verify the flowrates for the Control Room Emergency Ventilation System - Pressurization and the Auxiliary/Fuel Building Emergency Exhaust (Engineered Safety Features Ventilation System).

Section 50.36 of 10 CFR requires the TSs to be derived from the analyses and evaluations included in the safety analysis report and WCNOC states that these analyses conform to Regulatory Position 5.1.3. This means that the most conservative inputs (when a range of values existed) were used with the objective of determining a conservative postulated dose. However, the licensee appears to have used nominal values in the design basis radiological analyses and these values would not yield the most conservative doses.

1. For those accidents analyses where nominal flow rate values are used, please justify how WCNOC conforms to Regulatory Position 5.1.3 and if these analyses conform to RIS 2001-019,

   or

2. Please submit for the NRC staff's review revised radiological consequences analyses with the most restrictive values of plant parameters selected from the range of design values possible. Provide the inputs, assumptions, methodology technical basis for the analysis and the results of the analysis. Also, please justify the assumptions and inputs used in the analysis. Analysis inputs should be the most restrictive values of plant parameters selected from the range of design values allowed during operation and during the specific event so that the postulated consequences of the event are maximized.

RAI ARCB1-GENERAL-3 - All DBAs

RIS 2006-04, "Experience with Implementation of Alternative Source Terms," states, in part:

> An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable.

In Section 8, "NRC Regulatory Issue Summary 2006-04 Comparison," of Enclosure IV, WCNOC states that the submittal conforms to this RIS position.

Based on the information provided by the licensee, the NRC staff is unable to independently verify many of the proposed offsite, control room and the TSC doses for WCGS.

1. Please provide all inputs, assumptions and methods used for these calculations that were not previously provided. Also, the licenee is requested to include the inputs and outputs for the RADTRAD code for the staff's review.

PRA Operations and Human Factors Branch (APHB)

APHB-1 CREA – Also for Loss of Non-Emergency AC Power Analysis, Locked Rotor

RG 1.183, "Regulatory Position 5.1.2, "Credit for Engineered Safety 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.

Enclosure IV, Table A, "Conformance with Regulatory Guide 1.183 Main Sections," the licensee states that the WCGS analysis conforms to RG 1.183, Regulatory Position 5.1.2.

RIS 2006-04, "Experience with Implementation of Alternative Source Terms," states, in part:

> An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review, by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable.

The table entitled "NRC Regulatory Issue Summary 2006-004 Comparison," in Enclosure IV, states that the analysis conforms to the above RIS 2006-04 position.

The licensee's response to ARCB-RAI-3 in Enclosure VII to the letter dated January 17, 2017 states, in part:

> No new operator actions have been credited to support the AST methodology.

Enclosure IV, Section 4.3.6.2.2.2, "Secondary Releases," states, in part:

> The minimum SG water mass is increased after 2 hours to take credit for operators maintaining level at narrow range just on span as justified by the Emergency Management Guidelines (EMGs). The mass used in the analysis after 2 hours was calculated at 0% narrow range to appropriately bound the just on span level of 6% narrow range.

Based on Table 4.3-9, "Assumptions Used for Rod Ejection Analysis," WCNOC is proposing to change the "plant total SG minimum mass" credited in the Control Rod Ejection analysis after 2 hours, where it previously remained the same both before and after 2 hours.

The updated mass is based on the operators maintaining the steam generator level on-span for a reasonable time following an event.

The proposed change to use an updated mass appears to be dependent upon a newly credited operator actions since these actions were not previously credited in the radiological accident analyses. Therefore, the NRC staff requests the following information:

1. Please justify the revised mass after 2 hours for the minimum SG water mass. Also, please make available any calculations for the NRC staff's review and provide a description of the assumptions, inputs, and methodology used in the calculation.

2. Please provide a human factors assessment of the operator actions credited to maintain the SG level on-span. Information in support of the assessment may be found in NUREG-0711, Revision 3, "Human Factors Engineering Program Review Model" (ADAMS Accession No. ML12324A013) and NUREG-0800, "Standard Review Plan," Chapter 18.0, "Human Factors Engineering" (ADAMS Accession No. ML16125A114) Attachment A, "Guidance for Evaluating Credited Manual Operator Actions."

APHB-2 Control Room Modeling – Several Accidents

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 <u>are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures</u>. 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.

Enclosure IV, Section 4.3.2.1, "Control Room Model," of the submittal states, in part:

> For events that rely solely on the control room air intake monitors for control room isolation, the unfiltered inleakage to the control room will continue to be associated with the normal mode air intake. This results in the modeling of the control room unfiltered inleakage with the normal mode atmospheric dispersion factors until either an SI actuation signal <u>or manual operator action completes the control room isolation</u>.

1. Please state which accidents credit manual operator action to complete the control room isolation and state the time assumed to perform the action.

2. If the credited assumed time was not previously reviewed and approved by the NRC, please justify the time assumed.

3. If the assumed time was previously reviewed and approved by the NRC staff, please state where this approval is documented.

4. Please provide a human factors assessment of the manual operator actions (not previously approved by the NRC staff) to complete the control room isolation. Information in support of the assessment may be found in NUREG-0711 and NUREG-0800, Chapter 18.0, Attachment A, "Guidance for Evaluating Credited Manual Operator Actions."

Technical Specifications Branch (STSB)

STSB-1

The regulations in 10 CFR 50.36(c)(2) require that technical specifications include LCOs. LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When LCOs are not met, the licensee shall shut down the facility or follow the remedial action permitted by the technical specifications. The regulations at 10 CFR 50.36(c)(2)(ii) state that LCOs must be established for each item meeting one of four criteria:

> Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The Final Policy Statement on TSs Improvements for Nuclear Power Reactors, dated July 22, 1993, is provided at 58 FR 39132. It established a set of objective criteria and guidance for determining which regulatory requirements and operating restrictions should be included in the Technical Specifications. These criteria were later codified in 10 CFR 50.36 as described above.

In Enclosure VII to the letter dated January 17, 2017, WCNOC provided responses to the NRC staff's RAIs associated with a previous application on the same subject. Specifically, WCNOC provided responses to ARCB-RAI-21. The response states that the requirement for the decay time prior to fuel movement is not specified in the TSs, and that this requirement was removed in License Amendment No. 123, which was the conversion to the improved Technical Specifications.

Prior to the conversion to the improved TSs (ITSs), the WCGS TSs contained an LCO specifying that the reactor be subcritical for 100 hours before there is movement of irradiated fuel in the reactor core. Based on the safety evaluation for Amendment No. 123, the screening criteria for including the requirements in the ITS have been satisfied for Criterion 2 [of 10 CFR 50.36(c)(2)(ii)] since decay time is consistent with the assumptions used in an accident analysis. However, the activities necessary to be performed at WCGS before commencing movement of irradiated fuel ensure that 100 hours of subcriticality will elapse before there is movement of irradiated fuel in the core.

The NRC staff's approval to relocate this LCO to a licensee controlled document was based, in part, on the staff's understanding that the activities necessary to be performed at WCGS before commencing movement of irradiated fuel ensured that 100 hours of subcriticality will elapse before there is movement of irradiated fuel in the core. The safety evaluation concluded that an LCO was not required to ensure 100 hours have elapsed prior to fuel movement, and that sufficient regulatory controls exist under 10 CFR 50.59 to maintain the effect of the provisions in this specification. Therefore, even though Criterion 2 of 10 CFR 50.36(c)(2)(ii) was satisfied, the staff concluded that an LCO was not necessary.

The reduction in minimum decay time to 76 hours does not appear to be consistent with the statement that a minimum decay time of 100 hours would elapse before fuel movement would commence. It is not clear how the basis for the NRC staff's approval of the relocation of the decay time TS is being maintained.

1. Please provide an explanation of how the change in minimum decay time maintains the effect of the provisions in the decay time TS.

SUBJECT: WOLF CREEK GENERATING STATION – REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE AMENDMENT REQUEST FOR TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSES INCLUDING ADOPTION OF ALTERNATE SOURCE TERM (CAC NO. MF9307; EPID L-2017-LLA-0211) DATED DECEMBER 4, 2017

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DATE	12/1/17	12/1/17	11/1/17	10/31/17
OFFICE	NRR/DRA/ARCB/BC*	NRR/DORL/LPL4/BC	NRR/DORL/LPL4/PM	
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