WOLF CREEK
NUCLEAR OPERATING CORPORATION

Jaime H. McCoy
Vice President Engineering

June 19, 2018

ET 18-0018

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

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

2) Letter dated December 4, 2017, from B. K. Singal, USNRC, to A. C. Heflin, WCNOC, "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)

3) Letter ET 18-0012, dated April 19, 2018, from J. H. McCoy, WCNOC, to USNRC

Subject: Docket No. 50-482: Supplement to License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety Analysis Including Adoption of Alternative Source Term

To Whom It May Concern:

Reference 1 provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS). The proposed amendment would support transition to the Westinghouse Core Design and Safety Analysis methodologies. In addition, the amendment request included revising the WCGS licensing basis by adopting the Alternative Source Term radiological analysis methodology in accordance with 10 CFR 50.67, "Accident Source Term." Reference 2 provided a request for additional information (RAI) related to the application. Reference 3 provided responses to a portion of the RAI related to the application. It was determined that additional time was needed to provide the remaining responses to Reference 2. On April 26, 2018, WCNOC personnel contacted B. K. Singal, NRC Project Manager, and provided notification that additional time was required. This letter provides the remaining responses to the Reference 2.

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

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Attachment I provides the non-proprietary response to the RAI. Attachment II provides the proprietary response to the RAI. As Attachment II contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse Electric Company LLC, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 2.390 of the Commission's regulations. This affidavit, along with Westinghouse authorization letter, CAW-18-4758, Revision 0, "Application for Withholding Proprietary Information from Public Disclosure," is contained in the Enclosure.

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

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

Sincerely,

Jaime H. McCoy

Jaime H. McCoy

JHM/rlt

Attachments: I Response to Request for Additional Information (Non-proprietary)
II Response to Request for Additional Information (Proprietary)
III Proposed Technical Specification Change (Mark-up)
IV Revised Technical Specification Pages

Enclosure: CAW-18-4758, Revision 0, "Application for Withholding Proprietary Information from Public Disclosure"

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

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Jaime H. McCoy, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Jaime H McCoy
Jaime H. McCoy
Vice President Engineering

SUBSCRIBED and sworn to before me this 19th day of June , 2018.

RHONDA L. TIEMEYER
Notary Public, State of Kansas
My Appointment Expires
January 11, 2022

Rhonda L. Tiemeyer
Notary Public

Expiration Date January 11, 2022

Enclosure: CAW-18-4758, Revision 0, "Application for Withholding Proprietary Information from Public Disclosure"
(7 pages including cover sheet)

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CAW-18-4758

June 12, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: SAP-18-45, P-Attachment, "Supplemental 90 Day Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies [Proprietary]," June 2018

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-18-4758 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Wolf Creek Nuclear Generating Station.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-18-4758, and should be addressed to Edmond J. Mercier, Manager, Fuels Licensing and Regulatory Support, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 256, Cranberry Township, Pennsylvania 16066.

Edmond J. Mercier, Manager
Fuels Licensing and Regulatory Support

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Edmond J. Mercier, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 6/18/2018

Edmond J. Mercier, Manager
Fuels Licensing and Regulatory Support

(1) I am Manager, Fuels Licensing and Regulatory Support, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in SAP-18-45, P-Attachment, "Supplemental 90 Day Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies [Proprietary]," June 2018, for submittal to the Commission, being transmitted by Wolf Creek Nuclear Generating Station letter. The proprietary information as submitted by Westinghouse is that associated with Westinghouse Alternate Source Term analysis and Methodology Transition, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to support Wolf Creek for the Alternate Source Term analysis and Methodology Transition.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of Alternate Source Term analysis and Methodology Transition.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Attachment I: Response to Request for Additional Information (Non-proprietary)
(49 pages)

Letter ET 17-0001, dated 01/17/2017, provided the Wolf Creek Nuclear Operating Corporation (WCNOC) application to revise the Wolf Creek Generating Station (WCGS) Technical Specifications (TS) to the Nuclear Regulatory Commission (NRC). The proposed change replaces the WCNOC methodology for performing core design, non-loss-of-coolant-accident (non-LOCA), and LOCA safety analyses (for Post-LOCA Subcriticality and Cooling only) to the standard Westinghouse methodologies for performing these analyses, and associated TS changes. This application would also revise WCGS's TS and 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." Subsequently, WCNOC received Requests for Additional Information (RAI) related to this application from the Radiation Protection & Consequence Branch (ARCB) and the Nuclear Performance and Code Review Branch (see ADAMS Accession No. ML17331A178 and ML17265A014, respectively), among others. Responses to these RAIs were submitted in letters WO 18-0004 and ET 17-0024, dated 01/15/2018 and 10/18/2017, respectively. Subsequent to responding to the NRC RAIs, an audit of the submittal was conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756). As a result of this audit, supplemental information for the following ARCB RAI responses originally transmitted in Letter WO 18-0004 is provided below:

ARCB1-LOAC-1
ARCB1-LOAC-2
ARCB1-LLBA-2
ARCB1-LLBA-4
ARCB1-LOCA-1
ARCB1-LOCA-3
ARCB1-LOCA-5
ARCB1-FHA-2
ARCB1-FHA-3
ARCB1-FHA-5
ARCB1-FHA-6
ARCB1-SGTR-2
ARCB1-SGTR-5
ARCB1-SGTR-6
ARCB1-LRA-1
ARCB1-MSLB-1
ARCB1-MSLB-2
ARCB1-WT-4
ARCB1-WT-5
ARCB1-CREA-1
ARCB1-CONTROL ROOM-3
ARCB1-CONTROL ROOM-4
ARCB1-CONTROL ROOM-6
ARCB1-GENERAL-2
ARCB1-GENERAL-3

Additionally, during the final peer review of the RAI responses provided to the Nuclear Performance and Code Branch, an additional concern was identified with the Question 2 response provided in Letter ET 17-0024. This concern is also addressed below under the following header:

NRC ADAMS Accession No. ML17265A014 Question 2

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

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.*

Supplemental Response:

Following the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the LOAC event was re-analyzed to include a pre-accident iodine spike. At the audit, the NRC acknowledged the guidance is silent on pre-accident spikes for LOAC events, but noted that the Technical Specifications (TS) allow operation at pre-accident spike levels and that there must be a technical basis for that allowance. Thus, a pre-accident iodine spike was added to the LOAC dose analysis. For the pre-accident iodine spike case, it is assumed that a reactor transient has occurred prior to the LOAC and has raised the Reactor Coolant System (RCS) iodine concentration to the TS limit for a transient of 60 μCi/gm DE I-131 (i.e., 60 times the maximum equilibrium RCS iodine concentration). Additionally, doses for only the limiting 2-hour intervals are reported for the Exclusion Area Boundary (EAB). The LOAC doses are also affected by the responses to RAIs ARCB1-GENERAL-2 and ARCB1-GENERAL-3. The results are summarized in the response to ARCB1-GENERAL-3.

<u>RAI ARCB1-LOAC-2</u> *- Loss of Non-Emergency Alternating Current Power (LOAC)*

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.*

<u>Supplemental Response:</u>

The LOAC doses are also affected by the post-audit supplemental responses to RAIs ARCB1-LOAC-1, ARCB1-GENERAL-2 and ARCB1-GENERAL-3. As part of the reanalysis, the offsite dose limits discussed in the original response to ARCB1-LOAC-2 were applied and doses for only the limiting 2-hour intervals were reported for the EAB. The results are summarized in the response to ARCB1-GENERAL-3.

<u>RAI ARCB1-LLBA-2</u> *- Letdown Line Break Accident (LLBA)*

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.*

<u>Supplemental Response:</u>

Following the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the Letdown Line Break (LLB) event was re-analyzed to include a pre-accident iodine spike. At the audit, the NRC acknowledged the guidance is silent on pre-accident spikes for LLB events, but noted that the Technical Specifications (TS) allow operation at pre-accident spike levels and that there must be a technical basis for that allowance. Thus, a pre-accident iodine spike was added to the LLB dose analysis. For the pre-accident iodine spike case, it is assumed that a reactor transient has occurred prior to the LLB and has raised the Reactor Coolant System (RCS) iodine concentration to the TS limit for a transient of 60 μCi/gm DE I-131 (i.e., 60 times the maximum equilibrium RCS iodine concentration). Additionally, control room isolation (i.e. the initiation of emergency mode HVAC flows and filtration) was credited following a high radiation signal. The LLB doses are also affected by the responses to RAIs ARCB1-LLBA-4, ARCB1-GENERAL-2 and ARCB1-GENERAL-3. The results are summarized in the response to ARCB1-GENERAL-3.

RAI ARCB1-LLBA-4 - *Letdown Line Break Accident (LLBA)*

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 analysis input and output) to allow the NRC staff to confirm the values assumed.*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the NRC reviewer expressed a concern associated with the letdown line break flow rate assumed in the dose analysis. Specifically, the concern was associated with the fact that the letdown flow rate of 222 gpm was not doubled as currently described in Chapter 15.6.2 of the USAR to account for the backflow at the break site. Wolf Creek discussed that the letdown heat exchanger (directly downstream of the letdown orifices) would reduce the temperature of water downstream of the break below 212°F and prevent the water that backflows from flashing. However, the NRC reviewer was concerned that while the water would be below 212°F, Section 5.5 of Appendix A of Regulatory Guide 1.183 states that if the water is below 212°F, then the flashing fraction should be assumed to be 10%, unless a lower value can be justified. Thus, it was requested that either additional information be provided to justify not doubling the assumed break flow rate, or revise the letdown line break dose analysis with a doubled assumed break flow.

In regards to doubling the flow rate, while there are check valves downstream of the limiting letdown line break location, rather than quantifying the maximum water volume between the location of the check valves and the break location as well as validating the maximum possible leakage past the check valves, the flow rate modeled within the dose analysis has been doubled in order to account for any backleakage.

In addition to the concern of not doubling the flow rate, the validity of the 141 gpm value documented in the original response to ARCB1-LLBA-4 was also discussed. Specifically, the NRC reviewer agreed that due to the procedural limitations preventing a lineup with greater than 141 gpm from occurring, that 141 gpm was an appropriate value for use (assuming that it is subsequently doubled for the dose analysis).

Nevertheless, while procedural limitations preclude a lineup with a flow rate greater than 141 gpm, the most limiting lineup of 222 gpm is conservatively considered within the dose analyses. The purpose for using the limiting value of 222 gpm is to bound all possible configurations of the letdown system and to ensure that the analysis break flow rate exceeds the actual break flow rate.

Thus, the letdown line break dose analysis was revised to consider a break flow of 444 gpm in order to conservatively bound all possible configurations of the letdown system as well as any potential backflow through the break. The resulting doses, updated to address this item as well as the additional changes documented within this letter are summarized in the response to ARCB1-GENERAL-3.

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

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 stop spraying) 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.*

Supplemental Response:

Following the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the LOCA event was re-analyzed without credit for sedimentation. At the audit, the NRC challenged the past practice of crediting sedimentation without a plant-specific analysis of the post-accident aerosol distribution. Thus, the credit for sedimentation was removed from the LOCA dose analysis. To offset the impact on the doses, the credited spray duration was increased from 5 hours to 9.5 hours, consistent with the current licensing basis value of 9.55 hours. The LOCA doses are also affected by the responses to RAIs ARCB1-SGTR-6, ARCB1-CONTROL ROOM-3, ARCB1-GENERAL-2 and ARCB1-GENERAL-3. The results are summarized in the response to ARCB1-GENERAL-3.

RAI ARCB1-LOCA-3 *- Loss-of-Coolant Accident (LOCA)*

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 design basis accidents occur while the control room and control building envelope boundaries are open for the duration of the accident, the EES are not credited and have dose results that meets the limits in GDC 19 of 10 CFR 50 Appendix A 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 LCO 3.7.13 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). 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.*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the markups of LCO 3.7.13, "Emergency Exhaust System (EES)," provided within Attachment II of WO 18-0004 were discussed. These markups had been provided in order to address a previous concern with how an inoperable boundary impacts the EES and what responsive actions should be taken. The markups impose a condition upon LCO 3.7.13, "Emergency Exhaust System (EES)," similar to that of Condition B of LCO 3.7.10, "Control Room Emergency Ventilation System." Note that the current licensing basis for Wolf Creek is consistent with Standard Technical Specifications (STS). Namely, Condition B for LCO 3.7.10 of STS and Wolf Creek Technical Specifications (TS) require that mitigating actions immediately be implemented and verified to be acceptable within 24 hours. Whereas Condition B of LCO 3.7.13 of STS and Wolf Creek TS do not require the immediate implementation of mitigating actions but rather simply allow for 24 hours to return the building boundary to an operable status. Nevertheless, as previously stated, in order to address the previous concern associated with how the auxiliary building boundary is credited by the analysis, markups were developed and provided. These markups were based upon how Watts Bar had previously addressed a similar concern.

In regards to the markups, it was discussed that while the control room doses are addressed by verifying that main control room occupants do not exceed 10 CFR 50 Appendix A GDC 19 limits, the reviewer was concerned that the offsite doses had not been explicitly addressed by the markups. In order to address this concern, the NRC reviewer requested that either the TS markups be modified to address offsite doses, or explicitly determine the offsite doses with no credit for the EES.

To address this issue, the impact of not crediting the EES for the offsite doses was determined. Specifically, the only analysis that credits the EES for offsite doses is the LOCA doses calculation. Therefore, in order to address the impact of not crediting the EES for the offsite doses analysis, a scenario was analyzed with no credit for EES filtration. Regarding the location of the release, while the location of the EES exhaust is the unit vent, if the EES is not credited, the radioactivity will either leak out of the auxiliary building, remain within the building, or be exhausted from the unit vent. Thus, for the offsite doses, the release point will continue to be from the auxiliary building or the unit vent. Both of these locations are directly next to the containment structure. The χ/Q values for a release from sources located close to the containment structure are given in Table 4.1.1-24 of Enclosure IV of ET 17-0001. These values were utilized for the offsite doses case that did not credit the EES.

The offsite doses for the LOCA doses analysis (as previously stated, no other event credited EES filtration) were calculated with no credit for the EES. The resulting doses remained within regulatory limits. As offsite doses will remain within limits regardless of whether the EES is operable or not, no additional TS changes are required to verify that offsite doses do no exceed regulatory limits for Condition B of LCO 3.7.13 "Emergency Exhaust System (EES)."

RAI ARCB1-LOCA-5 - *Loss-of-Coolant Accident (LOCA)*

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 design basis accident and describe how these assumptions are considered in the radiological analyses. Note 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.*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the NRC reviewer expressed a concern that the time to isolate the Containment Isolation Valves had been removed from the Technical Specification (TS) Bases and that the replacement time had not been provided. Throughout the course of the discussions, it was determined that the information that had been identified for deletion on page B 3.6.3-2 of the TS Bases (Enclosure IV of ET 17-0001), should no longer be deleted. The paragraph of interest is shown below:

> The DBA analysis assumes that, after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. ~~The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.~~

The intent had been to markup Section B 3.6.3 of the TS Bases to be aligned with the mini-purge isolation time of 10 seconds. As the 10 second closure time is more limiting than the listed value of 60 seconds, the purpose of the change was to remove any potential to interpret that the mini-purge is isolated by 60 seconds rather than the more limiting time of 10 seconds. Note that the explicit time of 10 seconds was provided on the markups to page B 3.3.6-1 of the TS Bases.

However, during the audit, it was identified that the deleted text created confusion as it could be interpreted that no overall closure time existed for the remaining Containment Isolation Valves (i.e., valves other than the mini-purge isolation valves).

In order to address this concern, the text previously identified for deletion within the markup of page B 3.6.3-2 of Enclosure IV of ET 17-0001 will no longer be removed from the TS Bases. The resulting paragraph is subsequently shown:

> 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 containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

The mini-purge isolation time of 10 seconds will be listed in the TS Bases (as shown on page B 3.3.6-1 of the TS Bases markups), and the 60 second isolation time for the remaining Containment Isolation Valves will be retained on page B 3.6.3-2 of the TS Bases. The 60 second isolation time includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke time and is consistent with the current licensing basis. No change is being requested associated with the 60 second isolation value.

RAI ARCB1-FHA-2 *- Fuel Handling Accident (FHA)*

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.*

Supplemental Response:

Following the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the NRC expressed concerns that the conservatism inherent in the overall effective Decontamination Factor (DF) of 200 described in Regulatory Guide 1.183, Appendix B may not be preserved with fuel pressures up to 1500 psig, and thus, a lower overall DF is needed. To address the concern, the overall DF has been reduced.

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Thus, if a comparable level of conservatism to that built into Regulatory Guide 1.183 is applied, an overall DF of 170 is supported for fuel pin pressures up to 1500 psig. The analysis has been updated and the results are presented in the response to ARCB1-GENERAL-3.

<u>RAI ARCB1-FHA-3</u> *- Fuel Handling Accident (FHA)*

1. *Please provide the results of the evaluations 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.*

<u>Supplemental Response:</u>

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the NRC reviewer expressed a concern that the control room HVAC system is allowed to be inoperable during core alterations. The potential exists for an unirradiated fuel assembly to be dropped onto an irradiated assembly during core alterations. This would result in a fuel handling accident scenario in which the control room HVAC could not be credited. Therefore, it was requested that either an analysis be performed evaluating the resulting control room dose for the limiting event during core alterations, or add core alterations to the limits of applicability for the affected LCOs to preclude the event.

The NRC reviewer's concern is associated with an already identified generic issue (ADAMS Accession No. ML13246A358). Note that the generic issue is tied to TSTF-51. Adoption of TSTF-51 had been proposed in the original submittal, but was removed from the original amendment request due to the ongoing generic issue. This removal was discussed and agreed upon at the pre-submittal meeting for the revised license amendment request. Following the audit, as an additional effort to allow NRC approval of the revised LAR to move forward, WCGS proposed a license condition to adopt the ultimate resolution of the industry issue. Nevertheless, the NRC Staff felt that a license condition was not an acceptable approach pending generic resolution.

Therefore, in order to address this concern, markups adding core alterations to the limits of applicability for Technical Specifications 3.3.7, "Control Room Emergency Ventilation System (CREVS) Actuation Instrumentation" and 3.7.10, "Control Room Emergency Ventilation System (CREVS)" are provided in Attachments III and IV. Attachments III and IV provide the Proposed Technical Specification Changes (Mark-up) and Revised Technical Specification Page, respectively. These markups will ensure that the control room HVAC systems are operable during core alterations, precluding the need for additional dose analysis of the fuel handling accident. Specifically, the defined term, "CORE ALTERATIONS" will be added to each of the above TS LCO's Applicability.

Upon generic resolution, WCGS will evaluate further changes to the licensing basis.

RAI ARCB1-FHA-5 *- Fuel Handling Accident (FHA)*

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% of the free volume) and the fuel building on a case-by-case basis, but no guidance exists for mixing in the auxiliary building.*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the reviewer expressed a concern that additional justification was needed for several of the items modeled within the FHA in containment with an open personnel airlock analysis. As there is limited regulatory guidance and industry precedence for several of the parameters associated with this analysis, the reviewer wanted to ensure that sufficient conservatism exists. Specifically, additional justification was requested for the following parameters/assumptions:

Maximum Wind Speed

Termination of Inleakage into the Control Room Following a Control Room Ventilation Isolation Signal (CRVIS)

Auxiliary Building Volume

60 cfm Unfiltered Inleakage into Control Room Prior to a CRVIS

Additionally, at the audit it was discussed that the FHA in containment with an open personnel airlock originally discussed in the acceptance review supplemental response (ET 17-0011) had conservatively simplified the control room model by not modeling the control room equipment room (subsequently referred to as the equipment room). In order to more accurately model the FHA in containment with an open personnel airlock, the analysis has been revised to include the equipment room. Therefore, in addition to providing the original information requested at the audit, additional discussions associated with how the equipment room was modeled are also included. The inclusion of the effects of the equipment room in the model resulted in a decrease in calculated doses. Margin obtained from the more accurate model has been utilized to ensure additional conservatism in the remaining input parameters.

Addition of the Control Room Equipment Room

The location of the containment, auxiliary building, equipment room, and control room to one another is shown in Figure 1. As shown in the figure, radionuclides that exit containment through the personnel hatch must first enter the auxiliary building. Next, in order to ultimately reach the control room, radioactivity needs to travel from the auxiliary building to the equipment room. Once inside the equipment room, radionuclides may either leak or be transferred (via the control room HVAC system) into the control room.

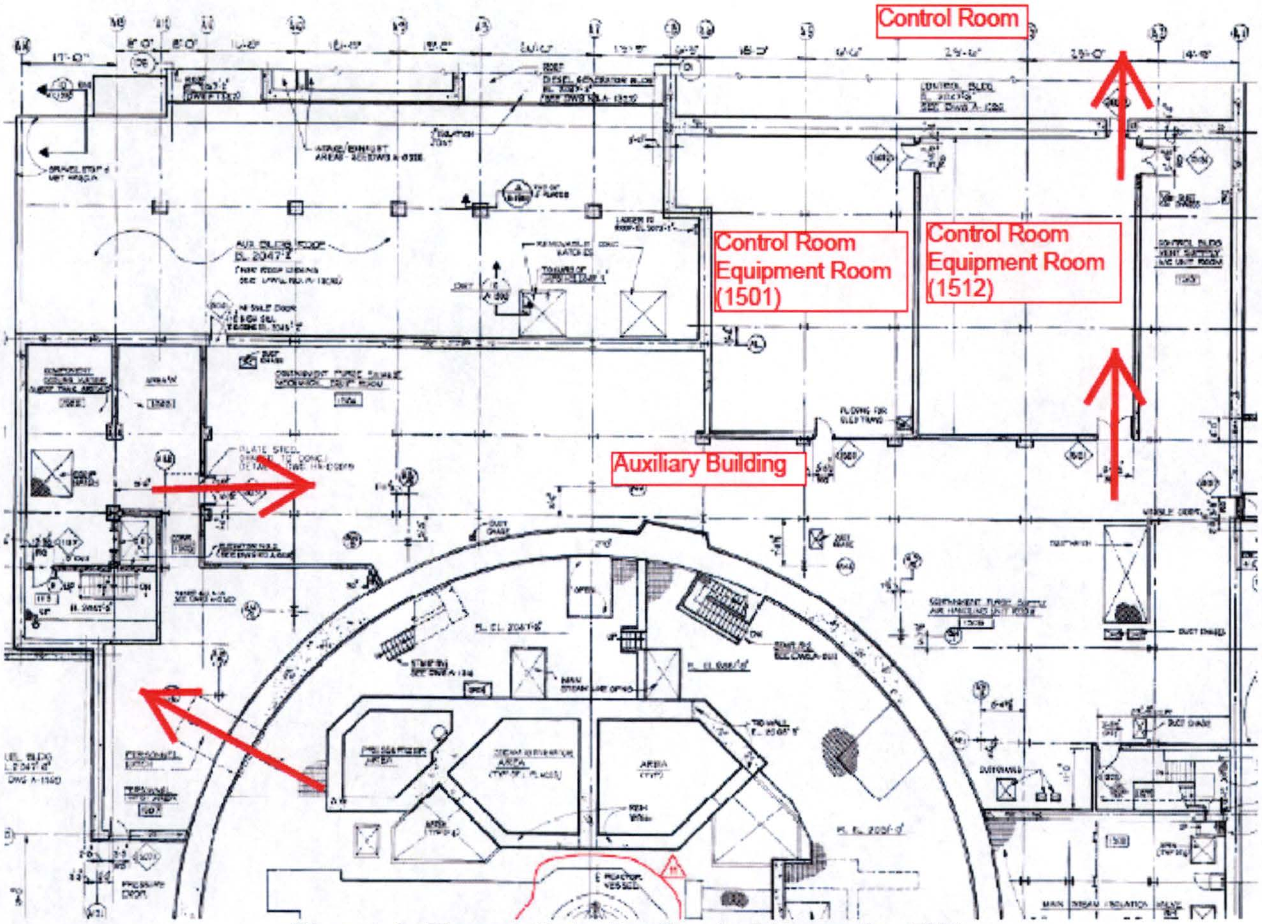

Figure 1: Flow path to the Control Room for FHA

The equipment room is part of the Control Room Envelope (CRE) and is serviced by the control room HVAC equipment. As shown on Figure 1, each train of control room HVAC equipment has its own independent equipment room. If both equipment rooms are considered, it will result in a larger volume available for dilution of the radionuclides. Furthermore, Room 1512 will have greater inleakage into it (from the auxiliary building) and from it (into the control room). This is due to the fact that there are two doors that connect Room 1512 to the auxiliary building (as opposed to one door for Room 1501) and one door that connects Room 1512 to the control room (as opposed to no doors connecting Room 1501 to the control room). Thus, in order to conservatively minimize the equipment room volume only the more limiting equipment room, 1512, is modeled.

The equipment room (1512) volume was calculated to be greater than 48,000 ft^3 by utilizing design drawings. As previously stated, the equipment room is part of the CRE and is serviced by the control room HVAC equipment during both normal and emergency modes of operation. The volume was rounded down to 30,000 ft^3 for conservatism and to account for equipment within the room.

During the normal mode of operation, the equipment room is supplied with 400 cfm of makeup air (from normal control room HVAC intake) and no air is mechanically exhausted from the room (excess air leaks out from the equipment room). During the emergency mode of operation, the equipment room is supplied with 350 cfm of filtered air from a combination of the control building and the control room and 300 cfm is filtered and then transferred to the control room.

The flowrates listed (400 cfm in during normal operation, 350 cfm in during emergency operation, and 300 cfm out during normal operation) are design flow rates. 10% uncertainty was applied to the flow rates for use in the analysis. As the supply flow rates reduce the overall dose by purging the equipment room, the flow rates were reduced by 10%. For the flow rate from the equipment room to the control room (300 cfm), if the net +50 cfm to the equipment room is modeled, it will result in a less limiting event as any excess flow rate to the equipment room results in a greater purging rate of the equipment room. Thus, while the equipment room is maintained at a positive pressure by a supply flow rate that this greater than the discharge flow rate, the analysis does not credit the purging to the environment. This is done by setting the equipment room to the control room flow rate (nominal 300 cfm) equal to the supply flow rate (350 cfm -10%) rather than crediting that the excess supply flow would be discharged to the environment.

<u>Maximum Wind Speed</u>

During the AST Methodology Transition audit, the NRC reviewer expressed a concern that the equation utilized to calculate the pressurization due to wind (obtained from ASCE 07-05) is based off of three second gusts and the wind speed utilized was based off of an hourly average. Thus, the reviewer asked that either the wind speed be converted from hourly averages to three second gusts, or provide justification that the hourly average values are appropriate.

In regards to the three second gusts, it is agreed that this approach would bound the maximum wind speed. However, while the analysis needs to consider a conservative wind speed, the three second gust period represents less than 0.2% of the thirty minute time frame. Thus, rather than converting the hourly average to a three second gust (less than 0.2% of the thirty minute timeframe), the hourly average is converted to a 10 minutes (600 second) timeframe as it is still bounded by the thirty minute timeframe. From Table 1.1 of WMO/TD-No. 1555, an hourly average wind speed is converted to a 10 minute duration by using a factor of 1.08 (WCGS is considered to be an In-Land site).

While converting the hourly average to a time period of 10 minutes addresses the concern that the average duration is not bounded by the event time frame of interest (30 minutes to manually actuate a CRVIS), it is recognized that the formula obtained from ASCE 07-05 may no longer be applicable. As such, the wind pressurization due to wind speed is recalculated by applying Bernoulli's theorem (base equation is subsequently provided).

$$\frac{p_1}{\rho_1} + \frac{v_1^2}{2g} + h_1 = \frac{p_2}{\rho_2} + \frac{v_2^2}{2g} + h_2 + h_L$$

Where:

p_1	=	pressure of air directly outside containment equipment hatch
ρ_1	=	density of air directly outside containment equipment hatch
v_1	=	velocity of air directly outside containment equipment hatch
h_1	=	height of air directly outside containment equipment hatch
p_2	=	pressure of air directly inside containment equipment hatch
ρ_2	=	density of air directly inside containment equipment hatch
v_2	=	velocity of air directly inside containment equipment hatch

h_2 = height of air directly inside containment equipment hatch
g = gravitational constant
h_L = head loss through equipment hatch

Where the head loss is given by

$$h_L = K\frac{v_1^2}{2g}$$

Where:

K = velocity head loss

The location of the two points (subscripts 1 and 2) are considered to be directly outside the equipment hatch and directly inside the equipment hatch. In order to maximize the pressurization due to wind, the velocity at point two is set to zero (due to the wind impacting solid structures inside of containment). Also, as the intent of the calculation is to determine the pressurization due to wind, and as the two locations are directly outside the equipment hatch and directly inside the containment hatch, the elevation difference between the two points is zero. Finally, in order to further simplify the equation, it is assumed that the density difference directly outside the equipment hatch and directly inside the equipment hatch is negligible. This is a reasonable assumption as 1) there is no heat source at the equipment hatch to impact the temperature, and 2) the pressure increase (subsequently determined to be 0.11 in. of water) has negligible impact on density. The simplified equation is therefore:

$$p_2 - p_1 = \frac{\rho_1 v_1^2}{2g}(1-K)$$

As documented within the previous response to ARCB1-FHA-5, an hourly average wind speed of 8.0 m/s (17.9 mph) bounds 95% of wind speeds that could pressurize containment. Additionally, as discussed within this response, the hourly average wind speed is converted to a 10 minute average by utilizing a factor of 1.08. For the density, as a maximum density is conservative relative to pressure increase, a value of 0.092 lbm/ft^3 is modeled (corresponds to the minimum site temperature of -30°F as discussed in Section 2.3.2.3 of the USAR). For the loss coefficient, the entrance into the containment via the equipment hatch is modeled as a flush pipe entrance with an r/d of 0.0, which results in a K of 0.5 (Crane Technical Paper 410). Therefore, the resulting wind pressurization is as follows:

$$p_2 - p_1 = \frac{\rho_1 v_1^2}{2g}(1-K)$$

$$p_2 - p_1 = \frac{0.092\,\frac{lbm}{ft^3}\left(1.08 * 17.9\,\frac{miles}{hour} * \frac{1\,hour}{3600\,sec} * \frac{5280\,ft}{1\,mile}\right)^2}{2 * 32.2\,\frac{lbm-ft}{lbf-sec^2}}(1-0.5)$$

$$p_2 - p_1 = 0.574\,psf$$
$$p_2 - p_1 = 0.0040\,psi$$
$$p_2 - p_1 = 0.11\,inches\ water$$

Termination of Inleakage into the Control Room Following a CRVIS

For the analysis documented within ET 17-0011, it was assumed that all inleakage into the control room is terminated following a CRVIS. The basis for this assumption is that following a CRVIS, the CRE will be pressurized at 0.25 in. of water pressure relative to the outside atmosphere. Thus, as the CRE will be at a higher pressure than the auxiliary building (as documented within this response, the pressurization due to wind will be 0.11 in. of water pressure), the flow of air will be from the control room to the equipment room and then to the auxiliary building. This is further demonstrated by the design of the control room HVAC system. Specifically, the equipment room is maintained at a positive pressure by supplying a net +50 cfm to the room (350 cfm supplied and 300 cfm exhausted). In order to balance the +50 cfm flow, an equal amount will need to leak out of the equipment room, rather than into it.

Furthermore, as discussed in the response to ARCB1-CONTROL ROOM-6, the containment penetrations will be isolated within 2 hours of the event occurring. Once the isolation of containment has occurred, the wind pressurization will be terminated.

Although the inleakage via penetrations will be terminated following a CRVIS, the contribution from ingress and egress is maintained for the duration of the event. While it is not expected that personnel will need to utilize either the doors from the auxiliary building to the equipment room or the door from the equipment room to the control room following a FHA, in order to avoid having to place restrictions on the use of any of the doors following an event, it is conservatively assumed that all ingress and egress takes place through the most limiting doors. Thus, the inleakage from the auxiliary building into the equipment room following a CRVIS is modeled as 10 cfm and the inleakage from the equipment room to the control room following a CRVIS is set to 10 cfm.

Auxiliary Building Volume

During the AST Methodology Transition audit, it was discussed that while Regulatory Guide 1.183 lists that credit for dilution within containment should generally be limited to 50%, no explicit guidance exists for what percentage of the auxiliary building should be credited. Thus, it was requested that additional justification be provided for the auxiliary building mixing volume credited.

The auxiliary building volume was calculated to be approximately 178,000 ft^3. Out of the total volume, 40% of the volume, 70,000 ft^3, is modeled within the analysis. The 40% value is bounded by the 50% value provided for the containment within Regulatory Guide 1.183. Relative to the 50% value listed in Appendix B of Regulatory Guide 1.183, the auxiliary building will have similar mixing due to the following: 1) there is a very torturous path from the personnel hatch to the equipment room (as shown on Figure 1) which will limit the quantity of radionuclides that reach the equipment room whereas for containment, the pathway is simply from the pool surface directly to the equipment hatch, 2) The high exhaust flow rate will result in a high air turnover rate; approximately 6 air changes per hour [70,000 ft^3/6,750 ft^3/min],[1] and 3) in order for air to move from the auxiliary building to the equipment room, the wind will need to pressurize the auxiliary building which will further promote mixing.

[1] As discussed in the response documented in ET 17-0011, a maximum exhaust flow rate is more limiting than a minimum exhaust flow rate.

Due to the similarities discussed between the auxiliary building and containment, it is judged that modeling 40% of the auxiliary building volume for mixing is appropriate for this application given that it is more limiting than the 50% mixing value for containment listed in Regulatory Guide 1.183.

60 cfm Unfiltered Inleakage into the Control Room Prior to a CRVIS

The 60 cfm unfiltered inleakage previously modeled within the FHA in containment with an open personnel airlock was used for the simplified control room model and was developed to address the in series unfiltered inleakage from the auxiliary building to the equipment room and the unfiltered inleakage from the equipment room to the control room. Thus, as the model was updated to explicitly model the equipment room, the one value of 60 cfm is no longer used. Rather explicit values were developed for both the unfiltered inleakage from the auxiliary building to the equipment room and the flow from the equipment room to the control room.

Regarding the unfiltered inleakage from the auxiliary building to the equipment room, as shown in Figure 1, there are two doors that connect the auxiliary building to the equipment room. In addition to these two doors, there are additional penetrations that pass through the wall that connects the rooms. Thus, all penetrations that unfiltered inleakage could pass through are considered.

In order to calculate the unfiltered inleakage through the various penetrations, the methodology outlined in NAA-SR-10100, "Conventional Buildings for Reactor Containment," was utilized.[2] Specifically, Equation 11 of NAA-SR-10100 can be used to calculate the inleakage (in cfm) via penetrations.

$$q = AP + BP^{1/2} \qquad \text{[Equation 11 of NAA-SR-10100]}$$

Where:

A	=	cfm per unit leak path per in. of water pressure
B	=	cfm per unit leak path per in.$^{1/2}$ of water pressure
P	=	pressure in in. of water pressure

[2] NAA-SR-10100 is included within the list of Department of Energy Standards list documented within Appendix C of DOE-TSL-1-2007, "DOE Technical Standards List."

Figure A-4(2)b of NAA-SR-10100 shows the leakage rate for metal doors with sound insulation as a function of pressure. As shown on the plot, at 0.11 in. of water pressure, the leakage rate is 16 cfm for a door that pressure tends to open and 12 cfm for a door that pressure tends to close. The methodology outlined in NAA-SR-10100 to scale the standard door was utilized for the two doors that connect the auxiliary building to the equipment room. Additionally, the overall leakage value for each door was doubled in order to account for the fact that the doors of interest are double doors. This conservative doubling accounts for the leakage through a portion of the door seals and doubles the contribution through corners, joints, and the lock jamb sections.[3] The resulting inleakage for the two doors is 39 cfm and 23 cfm at 0.11 in. of water pressure.

Next, the various penetrations through the wall that connects the auxiliary building to the equipment room were considered. There are 29 penetrations into the equipment room of interest. Twelve of the penetrations are filled with Dow Corning Silicone Foam, nine are grouted closed, two consist of a boot seal, and six are HVAC penetrations that are welded penetrations. As documented within NAA-SR-10100, the leakage constants for silicone foam, grouting, and welded HVAC penetrations are very small ('A' and 'B' values on the order of 10^{-5}). While the boot seal could not be correlated to one of the items tested in NAA-SR-10100, an explicit maximum leakage value is listed in the associated specification and was thus utilized to calculate the seal leakage for the two boot seals. The total inleakage through all 29 penetrations was determined to be less than 0.01 cfm. Compared to the significantly larger flow rates through the doors, the inleakage through the penetrations is considered to be negligible.

In addition to the inleakage through doors and penetrations, the inleakage due to ingress and egress (10 cfm) is explicitly included in the overall value.

Therefore, the total inleakage from the auxiliary building to the equipment room was determined to be 72 cfm (39 cfm + 23 cfm + 0.01 cfm + 10 cfm). This value was conservatively rounded up to 100 cfm.

Next, for the unfiltered inleakage from the equipment room to the control room, as shown in Figure 1, one door connects the equipment room to the control room. In addition to the one door, there are also penetrations that connect the equipment room to the control room that are considered. Finally, the contribution of air leaking into the normal HVAC equipment is also addressed.

For the one door that connects the equipment room to the control room, it has a specified maximum leakage rate of 0.1 cfm per linear foot at 0.25 in. of water. The resulting inleakage for the door was determined to be 2.1 cfm (no credit for a pressure less than 0.25 in. of water was taken).

Similar to the penetrations that connect the auxiliary building to the equipment room, the penetrations between the equipment room and the control room are either filled with silicone foam (15), grouted (2), or welded HVAC (11) and therefore, similar to the auxiliary building to

[3] As discussed in NAA-SR-10100, the 'A' term is used for the seals. For the double doors, the value was scaled up to address the mid seal and thus doubling the overall flow double accounts for the mid (between the two doors) seal leakage. For the 'B' term (corners, joints, and lock jamb sections), doubling the overall flow accounts for double the corners, joints, and lock jamb sections, which would be the case for two doors versus one.

the equipment room penetrations, the combined leakage rate is low at 0.11 in. of water pressure (less than 0.01 cfm). Thus, the leakage rate from the equipment room to the control room via penetrations is negligible.

Next, two isolation dampers per train are used to isolate the control room HVAC equipment from the equipment room during normal operation. The design leakage of the dampers is 30 cfm at 6 inches of water. For the limiting lineup, in regards to maximum suction pressure (both trains of control room HVAC are in operation), if the wind pressurization of 0.11 in. of water pressure is considered (no credit for reduction in pressure from the auxiliary building to the equipment room), the resulting suction pressure at the isolation dampers will be less than 6 inches of water. Additionally, one of the isolation dampers per train is on the discharge of the fan unit and thus the direction of flow would be from within the HVAC ductwork to the equipment room. Thus, only one isolation damper is considered for each control room HVAC train. While only one equipment room volume is considered in the calculation (in order to minimize the mixing volume), inleakage past two isolation dampers (the suction side damper from each train) is conservatively modeled. While the concentration in Room 1501 will be less than Room 1512, it is conservatively assumed that all inleakage comes from the room with the higher concentration. Therefore, the total inleakage due to the isolation dampers is modeled as 60 cfm (30 cfm per isolation damper).

Finally, the inleakage due to ingress and egress (10 cfm) through the door connecting the equipment room to the control room is also accounted for. It is worth noting that minimal to no personnel traffic will go through the door of interest following a FHA. However, in order to avoid placing limitations on its use following an event, it is conservatively assumed that all ingress and egress is through the limiting door.

Thus, the resulting inleakage from the equipment room to the control room will be 72.1 cfm (2.1 cfm + 0.01 cfm + 60 cfm + 10 cfm). The resulting inleakage is conservatively rounded up to 100 cfm.

In summary, the total unfiltered inleakage from the auxiliary building to the equipment room prior to a CRVIS is modeled as 100 cfm. The total unfiltered inleakage from the equipment room to the control room prior to a CRVIS is modeled as 100 cfm.

<u>Updated Results</u>

Based upon the changes outlined within this response, and the supplemental response to ARCB1-FHA-6 documented within this letter, the model was updated and the results were recalculated. The resulting doses are summarized in the response to ARCB1-GENERAL-3.

It is recognized that little regulatory guidance exists for several input values for this analysis. As explicit guidance does not exist for the more calculation intensive input values (specifically the auxiliary building volume, the equipment room volume, the unfiltered inleakage from the auxiliary building into the equipment room prior to a CRVIS, and the unfiltered inleakage from the equipment room to the control room prior to a CRVIS), it was the intent to compare the values utilized for the Wolf Creek analysis to similar approved analyses. However, only one similar approved analysis could be located. The approved analysis is for the Farley Nuclear Plant (discussed within ADAMS Accession Number ML17159A847 and approved within ADAMS Accession Number ML17271A265). In terms of values, while the docketed information does not list the percentage of the auxiliary building credited, it is documented that an auxiliary building volume of 100,650 ft^3 is modeled. However, as the percentage of the auxiliary building volume credited is not provided, it is not directly applicable to compare the 100,650 ft^3 to the Wolf Creek value of 70,000 ft^3. For the unfiltered inleakage, the approved Farley analysis models 10 cfm throughout the duration of the event. When compared to the Farley Analysis, the Wolf Creek analysis is conservative to the approved value of 10 cfm as the Wolf Creek models both a 10 cfm unfiltered inleakage throughout the duration of the event and an additional inleakage amount based upon maximum inleakage via penetrations due to wind pressurization.

RAI ARCB1-FHA-6 - *Fuel Handling Accident (FHA)*

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 10 CFR Part 50, Appendix A and 10 CFR 50. 67. In addition, please provide the inputs, assumptions, methodology a technical basis for the analysis, and justify the assumptions used.*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the NRC reviewer expressed a concern that additional information was needed in regards to how the penetrations are modeled for the FHA in containment with an open personnel airlock analysis. The first concern was associated with assuming that the containment penetrations are isolated at two hours while the administrative note for LCO 3.9.4, "Containment Penetrations," does not require the penetrations to be isolated following an event. The second concern is associated with the release rate from containment to the outside environment. Specifically, while the analysis considers the maximum flow rate to the auxiliary building (based upon the physical limitations of the HVAC systems), the reviewer was concerned that no additional leakage (e.g., directly to the environment) was considered.

In regards to the first concern, as the FHA in containment with an open personnel hatch analysis credits that containment penetrations are isolated two hours after an event, proposed changes to LCO 3.9.4, "Containment Penetrations," are being provided to ensure that all containment penetrations are isolated consistent with the safety analysis. These changes are discussed further within the supplemental response to ARCB1-CONTROL ROOM-6 within this letter.

For the second concern, in addition to the release pathway to the auxiliary building, the containment leakage rate directly to the environment was increased to a value directly below the Control Room Ventilation Isolation signal (CRVIS) setpoint in order to maximize the amount of radioactivity brought into the control room without generating an automatic actuation of the control room emergency HVAC equipment. This is conservative as if the leakage rate was high enough to result in a CRVIS, the unfiltered inleakage to the control room at the onset of the event (prior to the 30 minute manual actuation of CRVIS) would significantly decrease.

Based upon the changes outlined within this response, and the supplemental response to ARCB1-FHA-5 documented within this letter, the model was updated and the results were recalculated. The resulting doses are summarized in the response to ARCB1-GENERAL-3.

RAI ARCB1-SGTR-2 - *Steam Generator Tube Rupture (SGTR)*

1. *Please justify how the SGTR conforms to* [RG 1.183, Appendix F, Regulatory Position 5.4 and RG 1.183 Regulatory Position 5.1.2]... *or revise the analysis to be consistent with them.*

Supplemental Response:

Following the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the SGTR dose analysis was revised to model the effects of a loss of offsite power concurrent with the SGTR. In follow-up clarification calls, the NRC expressed reservations with the arguments made in the supplemental response that were based upon the removal of arbitrary conservatism in the analysis. Thus, SGTR dose analysis was revised to model the effects of a loss of offsite power concurrent with the SGTR. The prior proposed analysis had modeled the loss of offsite power at 52 seconds as a consequence of the reactor trip. For the revised SGTR dose analysis, the transient mass releases were shifted earlier by 52 seconds, and the pre-trip releases were removed from the model. The initiation of emergency mode filtration was credited at 60 seconds, instead of the 120 seconds modeled previously. The 60 second value bounds process rack time (< 0.3 seconds), SSPS time (< 2 seconds), master/slave relay and diesel start time (< 12.2 seconds), and damper closure time (< 30 seconds). The remaining approximately 15 seconds is added for conservatism.

The revised SGTR input parameters, replacing the equivalent data from Table 4.3-11 of Enclosure IV of "Wolf Creek, License Amendment Request for the Transition to Westinghouse Core Design and Safety Analyses" (ADAMS Accession No. ML17054C103), are below:

Input Parameter	Value
Transient mass transfer data	
Non-flashed break flow (lbm)	
0 – 1050 seconds	43,129.9
1050 – 2850 seconds	88,387.2
2850 – 3450 seconds	32,991.2
3450 – 3794 seconds	18,224.8
3794 – 5103 seconds	61,523.0
5103 – 7475 seconds	41,166.4
Flashed break flow (lbm)	
0 – 1050 seconds	2,901.8
1050 – 2850 seconds	13,432.1
2850 – 3450 seconds	2,635.6
3450 – 3794 seconds	606.1
Steam released from ruptured SG (lbm)	
0 – 1050 seconds	27,469.2
1050 – 2850 seconds	149,850.8
2850 – 7475 seconds	0
7475 – 43,200 seconds	2530
Steam released from intact SGs (lbm)	
0 – 1050 seconds	69,877.5
1050 – 3450 seconds	0
3450 – 3794 seconds	94,307.4
3794 – 5103 seconds	130,799.9
5103 – 7475 seconds	98,156.3
7475 – 43,200 seconds	1,645,930

The SGTR doses are also affected by the responses to RAIs ARCB1-SGTR-6, ARCB1-GENERAL-2 and ARCB1-GENERAL-3. The results are summarized in the response to ARCB1-GENERAL-3. The additional discussion provided in the initial supplemental response to RAI ARCB1-SGTR-2 relating to the modeling of the control room was affected by the responses to ARCB1-SGTR-6 and ARCB1-GENERAL-2, and so has been updated and included in the response to ARCB1-GENERAL-3.

RAI ARCB1-SGTR-5 - *Steam Generator Tube Rupture (SGTR), Main Steamline Break (MSLB), and other accidents that assume DEX-133*

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 10 CFR 50 Appendix A 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 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).*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the NRC reviewer expressed a concern associated with the current completion time of 48 hours for Condition B. of LCO 3.4.16, "RCS Specific Activity." The specific concern expressed by the reviewer was that the dose analyses consider that the maximum RCS activity is aligned with the LCO 3.4.16, "RCS Specific Activity," limits, not at levels elevated above corresponding Technical Specifications (TS) limits. As WCGS is allowed to operate for up to 48 hours above the Xe-133 TS limit, the reviewer was concerned that the initial RCS activity modeled within the analysis does not address operating at elevated Xe-133 levels for up to 48 hours.

It is noted that WCGS had not requested any changes to LCO 3.4.16, "RCS Specific Activity," as part of this License Amendment Request. Rather, the 48 hour time had previously been extended from 6 hours as part of License Amendment 170 (ADAMS Accession No. ML062790364). However, the NRC reviewer's concern was associated with an already identified generic issue (ADAMS Accession No. ML16113A402) that has not yet been resolved. WCGS proposed a license condition to adopt the ultimate resolution of the industry issue to allow NRC approval of the revised LAR to move forward pending generic resolution. Nevertheless, the NRC Staff felt that a license condition was not an acceptable approach.

Therefore, in order to address this concern, proposed Technical Specification markups removing the 48 hour completion time from LCO 3.4.16, "RCS Specific Activity," have been provided in Attachments III and IV. Attachments III and IV provide the Proposed Technical Specification Changes (Mark-up) and Revised Technical Specification Page, respectively. These changes were modeled off of how Cook Nuclear Plant previously addressed this generic issue (ADAMS Accession No. ML16327A110).

Upon generic resolution, WCGS will evaluate further changes to the licensing basis.

RAI ARCB1-SGTR-6 *- Steam Generator Tube Rupture (SGTR)*

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).*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the NRC reviewer expressed a concern that the atmospheric dispersion factors (χ/Q) for unfiltered inleakage did not consider all sources of inleakage. Specifically, while the control room is primarily surrounded by the control building, the North wall of the control room is next to the communications corridor building (not part of the Control Building Envelope (CRE)). In order to address this concern, it was requested that additional information be provided to justify the conservatism of the current χ/Q values, or update them to account for potential inleakage between the communications corridor building and the control room.

In order to address this concern, the χ/Q value for the emergency mode unfiltered inleakage has been modified to account for potential inleakage into the control room from the communications corridor building. The basis for the new value considers inleakage from the penetrations and inleakage from ingress and egress into the control room.

First, the penetrations between the control room and communications corridor were reviewed to evaluate the potential for inleakage. Based upon the review, all penetrations between the control room and communications corridor wall are either grouted closed or filled with silicone foam. These penetration types have very low potential for inleakage and are also periodically inspected to ensure that the penetration seal does not degrade over time. Furthermore, during the emergency mode of operation, the control room will be pressurized so any leakage through the low leakage penetrations will be from the control room to the communications corridor. Thus, as any possible leakage will be from the control room to the communications corridor during the emergency mode of operation, no inleakage via penetrations between the control room and the communications corridor is modeled.

Second, in addition to penetrations, the main access door to the control room is located on the North wall of the control room and is therefore connected to the communications corridor building. As operators are allowed to utilize this door to access the control room following a radiological event, the impact on ingress and egress will be considered. However, operators could also access the control room through doors connecting the control room to the control building. Thus, in order to ensure that the unfiltered inleakage due to ingress and egress into the control room is from the more limiting (in terms of concentration) of the two possible locations, the higher χ/Q for the two locations (control building versus communications corridor) is utilized. As χ/Q values had not previously been calculated for the communications corridor HVAC Intake, new ones needed to be developed. Figure 2 shows the locations of the communications corridor Intake relative to the other sources and receptors modeled within the dose analyses.

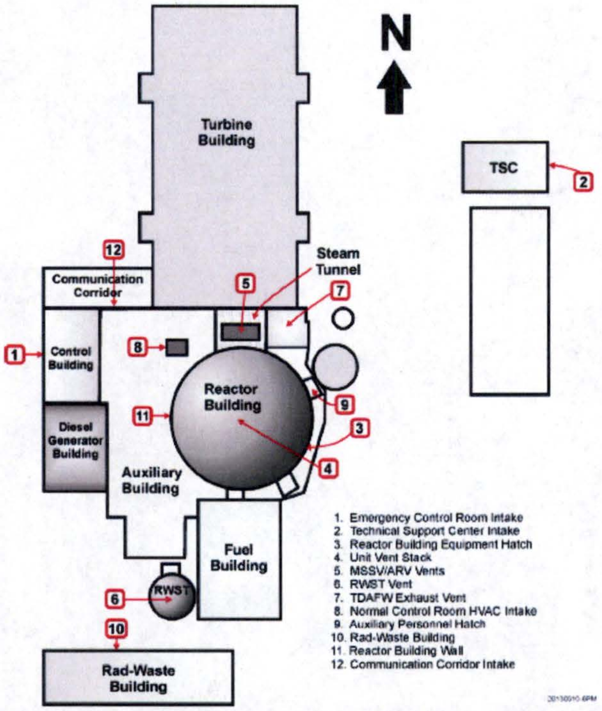

Figure 2: Communications Corridor Intake

As shown in the figure, the communications corridor intake is farther from the sources relative to the normal control room HVAC Intake, but closer relative to the emergency control room intake. The calculated χ/Q values also reflect this comparison, i.e., the communications corridor intake is more limiting than the emergency control room intake but less limiting than the normal control room HVAC intake. The χ/Q values calculated for the communications corridor are listed in Table 1 below:

Table 1: Communication Corridor Atmospheric Dispersion Factors χ/Q (sec/m3)

Time Period	Unit Vent Exhaust	MSSVs	RWST
0 to 2 hours	1.32E-03	5.10E-03	7.90E-04
2 to 8 hours	1.02E-03	3.77E-03	6.75E-04
8 to 24 hours	4.08E-04	1.43E-03	2.59E-04
1 to 4 days	2.99E-04	1.00E-03	2.20E-04
4 to 30 days	2.37E-04	7.34E-04	1.67E-04

Notes: The Unit Vent Exhaust χ/Qs are used for the LOCA and Rod Ejection Containment Leakage releases, LOCA Containment Purge, LOCA ECCS leakage to the auxiliary building releases, and MSLB faulted SG releases. The MSSV χ/Qs are used for the MSLB intact SG releases and the SGTR ruptured and intact SG releases. The RWST χ/Qs are used for the LOCA RWST back-leakage release. The other analyses do not credit closure of the normal HVAC intake damper.

Thus, for the normal HVAC mode of operation, the normal control room HVAC intake χ/Q value is retained for use as it is more limiting than the communications corridor intake. However, for the emergency HVAC mode of operation, the χ/Q has been modified to address the unfiltered inleakage due to ingress and egress. Specifically, after closure of the normal HVAC intake damper, the unfiltered inleakage is apportioned between the emergency mode HVAC intake (40 cfm) and the communications corridor intake (10 cfm). The 10 cfm is solely due to ingress and egress into the control room from the communication corridor (additional discussions on the modeling of ingress and egress provided in the supplemental response to ARCB1-Control Room-4 contained within this letter). The remaining 40 cfm out of the overall 50 cfm value is due to inleakage into the control room from the control building. The basis for modeling the control building (emergency mode HVAC intake) for the source of unfiltered inleakage other than that of ingress and egress is documented within the previous response to ARCB1-SGTR-6. Note that no credit for dilution within either the communications corridor or the control building has been taken for unfiltered inleakage within the analyses.

The resulting doses, updated to address this item as well as the additional changes documented within this letter are summarized in the response to ARCB1-GENERAL-3.

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

1. *Please clarify when the loss of offsite power is assumed and justify how this conforms to* [RG 1.183] *Regulatory Positions 5.1.2 and 5.4...*

Supplemental Response:

As a result of the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the NRC requested the following clarification to the previous response: The locked rotor dose analysis modeled the effects of a loss of offsite power at the start of the event. Releases from the steam generators to the atmosphere begin at the start of the event. This is conservative with respect to the calculated radiological consequences.

RAI ARCB1-MSLB-1 - *Main Steamline Break (MSLB)*

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.*

Supplemental Response:

As a result of the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the NRC requested the following clarification to the previous response: The main steamline break dose analysis modeled the effects of a loss of offsite power at the start of the event. Releases from the steam generators to the atmosphere begin at the start of the event. This is conservative with respect to the calculated radiological consequences.

RAI ARCB1-MSLB-2 *- Main Steamline Break (MSLB)*

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.*

Supplemental Response:

The MSLB doses are also affected by the post-audit supplemental responses to RAIs ARCB1-SGTR-6, ARCB1-GENERAL-2 and ARCB1-GENERAL-3. The results are summarized in the response to ARCB1-GENERAL-3.

RAI ARCB1-WT-4 *- Liquid Waste Tank Failure*

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

Supplemental Response:

As a result of the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the following clarifications are made for the Westinghouse Response to ARCB1-RAI-WT-4:

- F_{PCA} represents the fraction of RCS activity concentration that is entering the tank. The following values are used for the tanks in question:
  - Recycle Holdup Tank (RHUT) F_{PCA} = 1.0 based on the Safety Evaluation (SE) for Wolf Creek in NUREG-0881. The RHUT receives reactor coolant from letdown flow (shim bleed) and the reactor coolant drain tank (equipment drains).
  - Waste Holdup Tank F_{PCA} = 0.5 based on updates to the USAR in KMLNRC 82-175, Docket Number STN 50-482. The waste holdup tank receives reactor coolant from clean waste.
  - Floor Drain Tank F_{PCA} = 0.058 based on the SE for Wolf Creek in NUREG-0881 [Note that the flow to the Floor Drain Tank is defined as 'Dirty Wastes'. This designation does not refer to the radiological state of the waste, but does refer to the cleanliness of the drainage as it is assumed to capture debris from the room floors.]

- M represents the mass of water in the tank. Term $[\qquad\qquad]^{a,c}$, which defines the rate of activity entering the tank conservatively models M as the mass of the water in the tank, instead of the mass of the water in the RCS. For this application, the mass of water in the RCS is 2.20E+08 grams, which is larger than the mass of each individual tank.
  - RHUT mass = 2.12E+08 grams
  - Waste Holdup Tank mass = 3.78E+07 grams
  - Floor Drain Tank mass = 3.79E+07 grams

- DF represents the decontamination factor applied to the RCS activity entering the tank. A DF is only applied to the RHUT in order to model the recycle evaporator demineralizer which is upstream of the RHUT. From the SE for Wolf Creek in NUREG-0881, the iodine DF for the recycle evaporator demineralizer is 10, based on NUREG-0017, Revision 1, Table 1-4.

- The following statement is modified for clarification:

"At equilibrium, the term containing the exponential quantity $[\qquad\qquad]^{a,c}$ equals one..."

RAI ARCB1-WT-5 *- Liquid Waste Tank Failure*

1. *...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 is 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% release.*

Supplemental Response:

As a result of the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the following supplement is made for the Westinghouse Response to ARCB1-RAI-WT-5 in order to satisfy the NRC request for justification of a partition factor based on transient conditions. Based on the insufficient data available for justification, the partition factor is conservatively removed from the calculations:

The partition factor of 100 for the airborne iodine activity in the Volume Control Tank (VCT) is removed from the analysis. All iodine activity in the VCT is conservatively modeled to become airborne and is available for transfer to the Waste Gas Decay Tank (WGDT). Dose results for the WGDT accident were revised as indicated in the Supplemental Response to ARCB1-GENERAL-3.

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

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.*

Supplemental Response:

As a result of the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the NRC requested the following clarification to the previous response: The control rod ejection dose analysis modeled the effects of a loss of offsite power at the start of the event. Releases from the steam generators to the atmosphere begin at the start of the event. This is conservative with respect to the calculated radiological consequences.

RAI ARCB1-CONTROL ROOM-3 *- Control Room Dose*

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.*

Supplemental Response:

Following the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the control room dose from operator transit to and from the control room was recalculated. The NRC had challenged the applicability of the proposed breathing rate, which was the same as the Regulatory Guide 1.183 value. Thus, the breathing rate assumed was increased to 7.0E-04 m^3/sec (double the Regulatory Guide 1.183 value modeled in the original response), which is consistent with the approach utilized by the Farley Nuclear Plant (ADAMS Accession No. ML17159A847). Other changes to the LOCA analysis were incorporated into the releases models (as noted below).

The transit dose to the operator was evaluated to be 0.8 rem. Contributions to the dose occur from containment leakage, ESF system leakage, RWST back-leakage, direct dose from deposited radioactivity, and direct dose from activity remaining in containment. The LOCA doses are also affected by the responses to RAIs ARCB1-LOCA-1, ARCB1-SGTR-6, ARCB1-GENERAL-2 and ARCB1-GENERAL-3. The results are summarized in the response to ARCB1-GENERAL-3.

Additionally, it is noted that, in the original ARCB1-CONTROL ROOM-3 response, some of the control room dose components were mislabeled in the bullet lists. The transit dose was mislabeled as ""External Sources" and the external sources were mislabeled as "Containment Purge". The containment purge doses were not included in the bullet list, although they were included in the reported total. These are replaced by the doses reported in the response to ARCB1-GENERAL-3.

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

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.*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the NRC reviewer expressed a concern associated with how ingress and egress are accounted for within the analysis and corresponding test procedure. Specifically, the reviewer expected that the value for ingress and egress (10 cfm) would be explicitly accounted for rather than included within the overall unfiltered inleakage value. While it was discussed that Wolf Creek's surveillance test procedure allows for ingress and egress during the performance of the test, the reviewer expressed a concern that depending on how/when the test is performed, the ingress and egress into the control room may not bound the potential ingress and egress during accident conditions. Thus, it was requested that either additional information be provided to justify the inclusion of the 10 cfm within the overall test acceptance criterion, or change the test acceptance criterion or analysis input to a value that explicitly accounts for 10 cfm unfiltered inleakage due to ingress and egress.

In order to address this concern, the surveillance test procedure acceptance criterion (Procedure STS PE-061, "Control Room / Control Building Habitability Test") will be changed to a value to support an additional 10 cfm unfiltered inleakage due to ingress and egress. Specifically, in order to be aligned with the current analysis value of 50 cfm for the control room unfiltered inleakage, the surveillance test procedure acceptance criterion for control room unfiltered inleakage will be 40 cfm so as to allow for an additional 10 cfm unfiltered inleakage due to ingress and egress. The total value of 50 cfm is aligned with the input value utilized for the dose analyses.

In regards to implementation, the current licensing basis test acceptance criterion is 20 cfm for unfiltered inleakage into the control room. Thus, as the previous test results are less than or equal to 20 cfm, the test results support the future less limiting (in regards to the procedure acceptance criterion) AST value of 40 cfm during the timeframe prior to the next performance of the surveillance test procedure.

RAI ARCB1-CONTROL ROOM-6 *- Control Room Dose*

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).*

Supplemental Response:

During the AST Methodology Transition audit (documented by ADAMS Accession No. ML18107A756), the NRC reviewer expressed a concern that while Technical Specifications (TS) markups had been provided for LCO 3.7.10, "Control Room Emergency Ventilation System (CREVS)," with regards to clarifying the administrative controls, similar changes to additional LCOs may be required to ensure that the safety analysis is aligned with WCGS TS. Specifically, the following LCOs were identified:

LCO 3.6.3, "Containment Isolation Valves"
LCO 3.7.19, "Secondary System Isolation Valves (SSIVs)"
LCO 3.9.4, "Containment Penetrations"

Thus, it was requested that either TS markups for the associated LCOs be provided to ensure that the WCGS TS are aligned with the safety analysis or additional justification be provided to demonstrate that no additional TS markups are warranted.

LCO 3.6.3, "Containment Isolation Valves,"

Subsequent to the audit, the NRC held internal discussion and determined that no changes were needed to TS LCO 3.6.3 or it's associated Bases.

LCO 3.7.19, "Secondary System Isolation Valves (SSIVs)"

From the NRC SE approving license amendment 184 which created TS LCO 3.7.19 (current plant condition, emphasis added in **bold**):

The steam generator blowdown system is used to maintain the steam generator secondary side water chemistry within specifications. The blowdown system also provides the means to sample the secondary side of the steam generators, drain the steam generators during outages and re-circulate the steam generator water during wet layup conditions. The steam generator blowdown isolation valves

(SGBIVs) and steam generator blowdown sample isolation valves (SGBSIVs) are installed to prevent uncontrolled blowdown from more than one steam generator, thereby mitigating a DBA. The SGBSIVs also isolate the non-safety-related portions from the safety-related portions of the system. Open SGBIVs and SGBSIVs could create the possibility of an unisolated secondary side following a HELB. The open valves may also prevent the required flow of auxiliary feedwater to the intact steam generators following a HELB. Both of these situations invalidate the assumptions in the safety analyses for DBAs. Therefore, the SGBIVs and SGBSIVs should be subject to new TS requirements to ensure that assumptions in the safety analyses for DBAs remain valid.

The NRC staff reviewed the proposed new TS 3.7.19 and the licensee's justifications for the proposed changes. The NRC staff evaluated the proposed TS using the framework discussed at the end of Section 2.0 and the end of Section 3.0. The NRC staff determined that the LCO and applicability statements ***and notes*** *for the actions section are acceptable because they meet the requirements of 10 CFR 50.36(c)(2). The NRC staff also noted that the LCO and applicability statements and notes for the Actions section are consistent with TS sections for equipment with similar safety functions, such as main feedwater regulating valves (MFRVs) and MFRV bypass valves. In a letter dated March 12, 2009 (ADAMS Accession No. N1L090620129), the NRC staff requested additional justification for the 7-day completion time of Required Action A.1, since required action and completion times for valves with apparently similar safety functions, such as MFRVs and MFRV bypass valves, are 72 hours. The licensee provided their technical basis for the 7-day completion time by letter dated April 10, 2009. In that letter, the licensee stated that a failure of an MFRV or MFRV bypass valve has the potential to have a more significant effect on a main steam line break than a failure of an SGBIV or SGBSIV due to the difference in sizes of the piping and tubing for the respective lines. The piping associated with the MFRVs is 14 inches and the tubing associated with the SGBIV and SGBSIV is 3/8 inches. The licensee also stated that failure of an SGBIV has a negligible impact on the plant risk. The NRC staff reviewed the justification for the 7-day completion time and concluded that it provided an acceptable basis for the proposed 7-day completion time.*

The NRC staff concludes that the proposed changes listed in Section 3.0 meet the regulatory requirements specified in Section 2.0 of this safety evaluation. The proposed changes to TS 3.7.2 are more restrictive than the current WCGS TS requirements. The changes to TS 3.3.2 are editorial in nature, and addition of TS 3.7.19 will support continued safe operation of WCGS. In addition, the proposed changes will provide ***adequate assurance*** *that the necessary quality of systems and components is maintained, that facility operation will be* ***within safety limits****, and that the LCOs for the SGBIVs, SSIVs, and SGBSIVs will be met consistent with the requirements of 10 CFR 50.36. Therefore, the NRC staff concludes that the requested TS changes described in Section 3.0 are acceptable.*

In regards to the AST license amendment, the SGBIVs, SSIVs and SGBSIVs are treated in the same way as the current licensing basis; closed at the onset of the event.

Furthermore, the emergency response procedures verify that the SGBIVs, SSIVs, and SGBSIVs are isolated following receipt of a steam generator blowdown and sample isolation signal. Specifically, Attachment F of EMG E-0 (EMG E-0 is entered following a reactor trip or safety injection signal) is utilized to validate that the isolation valves have properly actuated. If a valve failed to isolate, then the valve will either be closed manually from the control room or locally isolated to ensure that there is not a pathway from the SG to the environment via the blowdown system.

Therefore, the current and revised SGTR event assumes no radioactive release from the SGBIVS, SSIVs, or the SGBSIVs. The NRC accepted that failure of an SGBIV (and thus, an SGBSIV) has a negligible impact on the plant risk in the current licensing basis. There is no change in plant risk due to the failure of an SGBIV or SGBSIV for the proposed licensing basis. The allowance for administrative controls in TS 3.7.19 is consistent with the allowance for administrative controls in TS 3.6.3 for containment isolation valves without direct access to outside atmosphere and thus no additional clarifications to the note are warranted.

LCO 3.9.4, "Containment Penetrations"

LCO 3.9.4 is for Containment Penetrations during refueling operations and thus the LCO is limited to the Fuel Handling Accident (FHA). The release model for the FHA described within Enclosure IV of ET 17-0001 is provided in Section 4.3.12.2.2.:

> *All activity released from the fuel pool is assumed to be released to the atmosphere in 2 hours using a linear release model. No credit is taken for filtration from the spent fuel pool ventilation system operation for the FHA in the fuel building. No credit is taken for isolation of containment for the FHA in containment. For these conditions, the assumptions and parameters for a FHA inside containment are identical to those for an FHA in the fuel building, and therefore, the radiological consequences are the same regardless of the accident location.*

Therefore, in regards to the analysis documented in Enclosure IV of ET 17-0001, as no credit is taken for isolating the containment within the analysis, the time to isolate the containment within two hours following a FHA is solely for defense in depth and is not required in order to meet the assumptions of the analysis.

However, as documented within the supplemental response to ARCB1-FHA-5 and ARCB1-FHA-6 (contained within this letter), the FHA within containment with an open personnel air lock credited isolation of containment at two hours. Thus, in order to support the assumptions contained within the FHA within containment with an open personnel air lock analysis, markups of TS LCO 3.9.4, "Containment Penetrations," have been provided in Attachments III and IV. Attachments III and IV provide the Proposed Technical Specification Changes (Mark-up) and Revised Technical Specification Page, respectively.

In summary, additional justification has been provided for why LCO 3.7.19, "Secondary System Isolation Valves (SSIVs)," is aligned with the safety analysis. Markups to LCO 3.9.4, "Containment Penetrations," that identify the changes to the administrative controls associated with unisolated penetrations have been provided. No additional justification was determined to be needed for TS LCO 3.6.3.

RAI ARCB1-GENERAL-2 - *Several Accidents*

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.*

Supplemental Response:

During the AST Methodology Transition audit (documented by documented by ADAMS Accession No. ML18107A756), it was discussed that the NRC reviewer was concerned with the use of nominal design flow rates for the control room HVAC equipment during the normal mode of operation.

These normal HVAC flow rates are modeled by all radiological accidents prior to the receipt of a Safety Injection Signal or a Control Room Ventilation Isolation Signal (CRVIS) (also includes the time to transfer to the HVAC emergency mode of operation). Nominal design flow rates had been modeled as the associated equipment is not safety-related. Thus, rather than crediting the equipment for the safe shutdown of the plant, the equipment is modeled in order to yield a more limiting event. Alternatively, as the safety-related control room HVAC equipment is credited to mitigate the event, uncertainty had been explicitly applied to the flow rates.

It is also noted that the nominal flow rates had previously been used for the postulated waste gas decay tank failure analysis which had supported License Amendment 200 (ADAMS Accession Number ML12318A145).

Nevertheless, in order to address this concern, it was requested that either additional information be provided to justify the use of nominal flow rates for the HVAC equipment during the normal mode of operation, or uncertainty be applied to the normal mode of operation flow rates in the appropriate direction.

To address this concern, uncertainty has been applied to the flowrates modeled for the HVAC equipment operating during the normal mode of operation. Consistent with the flowrate uncertainty listed in Section 5.5.11, "Ventilation Filter Testing Program (VFTP)" of WCGS Technical Specifications (TS), a 10% factor was applied to the normal mode flow rates. As filtration is not credited for the normal mode of operation, a higher flowrate will result in more radioactivity being brought into the control building and control room and thus the 10% factor was applied in the positive direction. The revised normal HVAC flow rates are as follows:

13050 cfm *1.1 = 14355 cfm, rounded to 14360 cfm.

1950 cfm makeup * 1.1+ 50 cfm inleakage = 2145 cfm + 50 cfm = 2195 cfm.

The resulting doses, updated to address this item as well as the additional changes documented within this letter are summarized in the response to ARCB1-GENERAL-3.

RAI ARCB1-GENERAL-3 *- Several Accidents*

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

Supplemental Response:

Following the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office (ADAMS Accession No. ML18107A756), the dose analyses were updated to address several RAIs:

ARCB1-LOAC-1 and ARCB1-LOAC-2: A pre-accident iodine spike was added to the dose analysis. Additionally, doses for only the limiting 2-hour intervals are reported for the EAB.

ARCB1-LLBA-2: A pre-accident iodine spike was added to the dose analysis. Note that control room isolation (i.e. the initiation of emergency mode HVAC flows and filtration) was credited following a high radiation signal.

ARCB1-LLBA-4: The flowrate assumed for the broken letdown line is increased from 222 gpm to 444 gpm to conservatively account for reverse break flow.

ARCB1-LOCA-1: Credit for sedimentation is removed. Note that the spray duration was increased from 5 hours to 9.5 hours (consistent with the Current Licensing Basis (CLB)) to offset the impact on the doses.

ARCB1-FHA-2: The overall pool decontamination factor for iodine was decreased from 200 to 170.

ARCB1-FHA-5 and ARCB1-FHA-6: The equipment room and associated HVAC flows were added to the model.

ARCB1-SGTR-2: The SGTR doses were reanalyzed to reflect a loss of offsite power at the start of the event. Note that control room isolation was credited at 60 seconds.

ARCB1-SGTR-6: The effects of the communication corridor HVAC intake on the unfiltered inleakage were added to the dose analyses.

ARCB1-WT-5: The iodine partition factor used in the calculation of the waste gas decay tank iodine inventory is removed.

ARCB1-CONTROL ROOM-3: The breathing rate applied to the transit doses is doubled to 7E-04 m^3/sec.

ARCB1-GENERAL-2: Normal mode control building and control room HVAC flows were conservatively increased by 10%. This applies to all dose analyses.

In addition, the increase (typically 10%) applied to the final calculated doses was removed.

The dose analyses were updated to reflect the responses to the above RAIs. It is noted that the analyses updates made prior to the audit (e.g. letdown line break airborne fraction [ARCB1-

LLBA-1], main steamline break control room isolation timing [ARCB1-MSLB-2], and reduced LOCA RWST back-leakage rate [ARCB1-CONTROL ROOM-4]) have been retained. Additionally, the discussion on control room modeling that was previously part of the supplemental response to ARCB1-SGTR-2 was updated to reflect the above changes and included below. The final calculated doses are presented at the end of this supplemental response in Table 5 and Table 6.

Control Room Isolation

The control room isolation is modeled in the dose analyses in two parts: actuation of the emergency mode filtration (in both the control building and the control room) and closure of the normal HVAC intake damper. The actuation of the emergency mode filtration occurs following receipt of an isolation signal (e.g. high radiation, safety injection or manual action). In the analyses, emergency mode filtration is actuated after a delay of at least 60 seconds to account for instrumentation delays and damper movement following an automatic isolation signal.

Closure of the normal HVAC intake damper occurs on a safety injection signal. It would also occur on manual action but this is not credited in the analyses (with the exception of the Fuel Handling Accident in containment with an open personnel hatch discussed in ARCB1-FHA-5 and ARCB1-FHA-6). The total unfiltered inleakage modeled during emergency mode is 50 cfm. Prior to the closure of the normal HVAC intake damper, the 50 cfm unfiltered inleakage is associated with the normal HVAC intake χ/Q. After closure of the normal HVAC intake damper, the unfiltered inleakage is apportioned between the emergency mode HVAC intake (40 cfm) and the communications corridor intake (10 cfm associated with ingress/egress). The χ/Qs associated with the communications corridor that are modeled in the analyses are presented in Table 2. See also the supplemental response to ARCB1-SGTR-6.

Table 2: Communication Corridor Atmospheric Dispersion Factors χ/Q (sec/m3)

Time Period	Unit Vent Exhaust	MSSVs	RWST
0 to 2 hours	1.32E-03	5.10E-03	7.90E-04
2 to 8 hours	1.02E-03	3.77E-03	6.75E-04
8 to 24 hours	4.08E-04	1.43E-03	2.59E-04
1 to 4 days	2.99E-04	1.00E-03	2.20E-04
4 to 30 days	2.37E-04	7.34E-04	1.67E-04

Notes: The Unit Vent Exhaust χ/Qs are used for the LOCA and Rod Ejection Containment Leakage releases, LOCA Containment Purge, LOCA ECCS leakage to the auxiliary building releases, and MSLB faulted SG releases. The MSSV χ/Qs are used for the MSLB intact SG releases and the SGTR ruptured and intact SG releases. The RWST χ/Qs are used for the LOCA RWST back-leakage release. The other analyses do not credit closure of the normal HVAC intake damper.

In the analyses, a failure of one of the filtration fans is assumed at the start of emergency mode resulting in a larger unfiltered inflow to the control room (since only half of the makeup flow to the control room passes through a filter). After a defined time of 90 minutes from the start of the event, operator action isolates the failed train and terminates the unfiltered inflow to the control room, and consequently lowers the filtered inflow to the control building.

A summary of control room modeling assumptions for all events is provided below:

Table 3: Control Room Modeling Assumptions

Event/Scenario	High Radiation Signal, Time from event initiation (sec)	SI Signal Generation, Time from event initiation (sec)	Emergency Mode Actuation Credited (sec)	Normal HVAC Intake Damper Closure (sec)
Main Steamline Break, both iodine spikes	N/A	Immediate	120 seconds*	120 seconds*
Loss of AC Power, both iodine spikes	N/A	N/A	N/A	N/A
Locked Rotor	Immediate	N/A	120 seconds	N/A
Control Rod Ejection – Containment Leakage	N/A	<150 seconds	210 seconds	210 seconds
Control Rod Ejection – Primary to Secondary Leakage	Immediate	N/A	120 seconds	N/A
Letdown Line Break , both iodine spikes	Immediate	N/A	120 seconds**	N/A
SGTR, both iodine spikes	Immediate	325 seconds	60 seconds	600 seconds
LOCA	N/A	Immediate	120 seconds	120 seconds
Tank Ruptures	N/A	N/A	N/A	N/A
Fuel Handling Accident	Immediate	N/A	120 seconds	N/A
Fuel Handling Accident – Auxiliary Building Releases	N/A	N/A	30 minutes (operator action)***	N/A

*See response to ARCB1-MSLB-2
**See supplemental response to ARCB1-LLBA-2
***See response to ARCB1-FHA-5

The HVAC flows are illustrated in Figure 3 with flowrates described in Table 4:

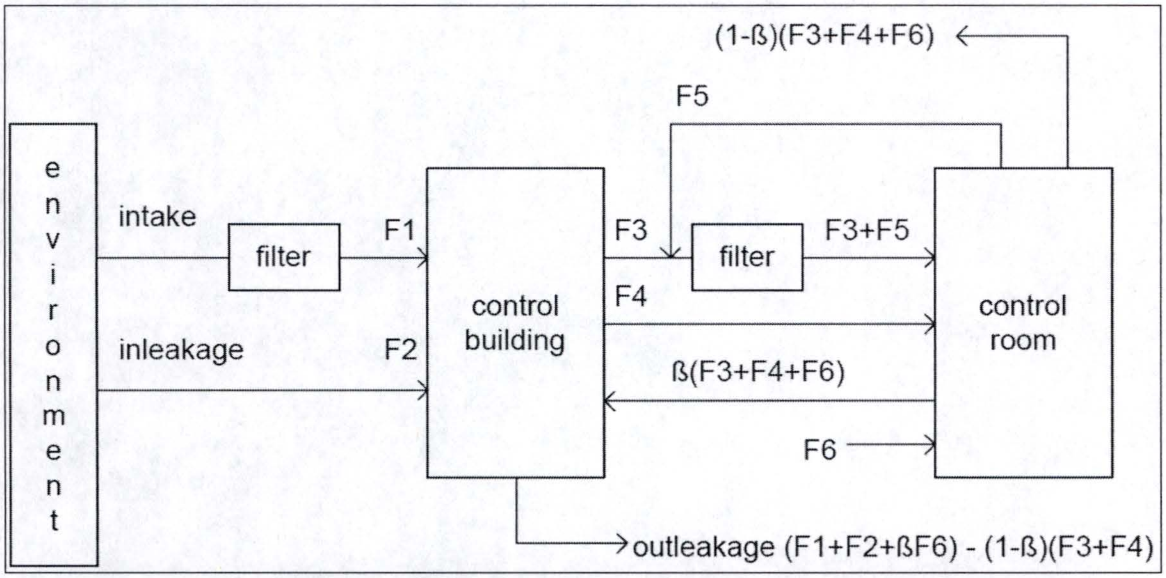

Figure 3: Control Room and Control Building Ventilation Flows

Table 4: Control Room and Control Building Ventilation Flows			
Flow Path	**Normal Mode Flow (cfm)**	**Emergency Mode Flow Prior to Operator Action (cfm)**	**Emergency Mode Flow After Operator Action (cfm)**
F1	0	1350	675
F2	14360*	400	400
F3	0	550	550
F4	0	550	0
F5	0	1250	1250
F6	2195**	50***	50***
β	0	0	0

*13050 cfm *1.1 = 14355 cfm, rounded to 14360 cfm.

** 1950 cfm makeup * 1.1+ 50 cfm inleakage = 2145 cfm + 50 cfm = 2195 cfm.

*** After Normal HVAC intake closure: 10 cfm via communications corridor associated with ingress/egress plus 40 cfm via Emergency HVAC intake

Revised Doses

The dose analyses were revised in response to the RAIs described above. The revised doses are tabulated below. These doses do not contain the adder (typically 10%) that was described in the previous response to ARCB1-GENERAL-3. Instead, the calculated total doses are rounded up to 2 significant figures. Table 5 contains the updated doses for events other than LOCA, and Table 6 contains the LOCA doses.

Table 5: Updated Doses (rem TEDE)				
Event/Location	**EAB***	**LPZ**	**CR**	**TSC**
MSLB – AI Spike	0.58	0.54	4.8	0.44
MSLB – Pre-Accident Spike	0.20	0.12	4.5	0.28
LOAC – AI Spike	0.0013	0.0047	2.5	0.0034
LOAC – Pre-Accident Spike	0.0018	0.0015	0.86	0.0021
Locked Rotor	0.38	0.32	3.5	0.16
CREA – Containment Leakage	1.1	1.9	2.6	2.0
CREA – Secondary Releases	0.38	0.32	3.5	0.16
LLB – AI Spike	0.35	0.12	0.37	0.43
LLB – Pre-Accident Spike	0.57	0.19	1.5	0.78
SGTR – AI Spike	0.80	0.26	1.1	1.5
SGTR – Pre-Accident Spike	0.99	0.32	4.2	2.2
LOCA	See Table 6			
Waste Gas Decay Tank Rupture	0.090	0.029	0.057	0.0076
Recycle Holdup Tank Rupture	0.025	0.0080	0.053	0.0058
Hypothetical Liquid Waste Tank Rupture	0.045	0.015	0.23	0.024
FHA	1.2	0.39	1.1	1.1
FHA – Auxiliary Building Releases	N/A	N/A	0.81	N/A

*The pre-accident spike scenarios added to the LOAC and LLB dose analyses have limiting 2-hour intervals for the EAB of 10 to 12 hours and 0 to 2 hours, respectively. The limiting 2-hour interval for the LOAC accident-initiated iodine spike is 10 to 12 hours. The limiting 2-hour intervals for the EAB dose for the other analyses are unchanged from those previously reported.

Table 6: LOCA				
Event/Location	**EAB**	**LPZ**	**CR**	**TSC**
Containment Leakage	4.36E+00	1.82E+00	1.09E+00	2.89E+00
ECCS Leakage	4.21E-01	1.15E+00	1.00E+00	5.60E-01
RWST Back-leakage	1.08E-03	3.40E-01	6.13E-01	1.46E-01
Containment Purge	0.0	7.50E-04	6.78E-02	3.57E-03
Transit*	N/A	N/A	8.0E-01	N/A
External Sources	N/A	N/A	1.26E-01	6.32E-01
Total	**4.8**	**3.4**	**3.7**	**4.3**

*The transit doses are discussed in the response to ARCB1-CONTROL ROOM-3.

<u>NRC ADAMS Accession No. ML17265A014 Question 2</u>

1. *Please provide further justification for the gap fractions assumed in the fuel handling accident analysis. The justification should provide an analysis using NRC-approved methodologies and a power history that bounds limiting plant-specific power histories at Wolf Creek, per RG 1.183 Footnote 11.*

Supplemental Response:

A clarification call was held on May 29, 2018 to discuss an additional concern that arose during the final peer review of the scope tied to the gap fractions. The concern is regarding the limitations on the quantity of fuel that may exceed the RG 1.183, Table 3, Footnote 11 conditions.

As described in the original response, typical Wolf Creek core designs show no exceedances of the Footnote 11 applicability limits. However, there is the possibility that an atypical core design (e.g. Cycle 23) could result in a limited number of rods (much less than 10%) exceeding the Footnote 11 applicability limits. Thus, an upper limit of 10% will be validated on a cycle-by-cycle basis as part of the reload safety analysis checklist.

Attachment III: Proposed Technical Specification Changes (Mark-up)
(8 pages)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time for Condition A, B or C not met in MODE 1, 2, 3, or 4.	D .1	Be in MODE 3.	6 hours
	AND		
	D .2	Be in MODE 5.	36 hours
E. Required Action and associated Completion Time for Condition A, B or C not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	E.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E .2	Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

---NOTE---

Refer to Table 3.3.7-1 to determine which SRs apply for each CREVS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days

(continued)

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, ~~and (a)~~ (a) and (c)	2	SR 3.3.7.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, ~~and (a)~~ (a) and (c)	2 trains	SR 3.3.7.3	NA
3.	Control Room Radiation-Control Room Air Intakes	1, 2, 3, 4, ~~and (a)~~ (a) and (c)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.5	(b)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

(b) Trip Setpoint concentration value (μCi/cm^3) is to be established such that the actual submersion dose rate would not exceed 2 mR/hr in the control room.

INSERT HERE

(c) During CORE ALTERATIONS.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	--------------------NOTE-------------------- LCO 3.0.4c. is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 $\leq 60\ \mu Ci/gm$.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	~~NOTE~~ ~~LCO 3.0.4c. is applicable.~~	
	B.1 ~~Restore DOSE EQUIVALENT XE-133 to within limit.~~	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C~~B. Required Action and associated Completion Time of Condition A ~~or B~~ not met. OR DOSE EQUIVALENT I-131 > 60 μCi/gm.	~~C~~B.1 Be in MODE 3. AND ~~C~~B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	---------------------------NOTE--------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 500 μCi/gm.	7 days
SR 3.4.16.2	---------------------------NOTE--------------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

---NOTE---
The control room envelope (CRE) and control building envelope (CBE) boundaries may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, and 4,
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREVS train inoperable for reasons other than Condition B.	A.1	Restore CREVS train to OPERABLE status.	7 days
B. One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary in MODES 1, 2, 3, or 4.	B.1	Initiate action to implement mitigating actions.	Immediately
	<u>AND</u>		
	B.2	Verify mitigating actions to ensure CRE occupant radiological exposures will not exceed limits and CRE occupants are protected from chemical and smoke hazards.	24 hours
	<u>AND</u>		
	B.3	Restore CRE boundary and CBE boundary to OPERABLE status.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	D.1 Place OPERABLE CREVS train in CRVIS mode. OR D.2.1 Suspend CORE ALTERATIONS. AND D.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately
E. Two CREVS trains inoperable during movement of irradiated fuel assemblies. OR One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary during movement of irradiated fuel assemblies.	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

(continued)

or during CORE ALTERATIONS

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts, or if open, capable of being closed;

b. One door in the emergency air lock closed and one door in the personnel air lock capable of being closed; and

---NOTE---
An emergency personnel escape air lock temporary closure device is an acceptable replacement for an emergency air lock door.
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c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge Isolation valve.

---NOTE---
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

that ensure the building boundary can be closed consistent with the safety analysis

Attachment IV: Revised Technical Specification Pages
(8 Pages)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time for Condition A, B or C not met in MODE 1, 2, 3, or 4.	D .1	Be in MODE 3.	6 hours
	AND		
	D .2	Be in MODE 5.	36 hours
E. Required Action and associated Completion Time for Condition A, B or C not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	E.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E .2	Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.7-1 to determine which SRs apply for each CREVS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT.	92 days

(continued)

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1, 2, 3, 4, (a), and (c)	2	SR 3.3.7.4	NA
2.	Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1, 2, 3, 4, (a), and (c)	2 trains	SR 3.3.7.3	NA
3.	Control Room Radiation-Control Room Air Intakes	1, 2, 3, 4, (a), and (c)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.5	(b)
4.	Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

(b) Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 2 mR/hr in the control room.

(c) During CORE ALTERATIONS.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	----------NOTE---------- LCO 3.0.4c. is applicable. ------------------------------		
	A.1	Verify DOSE EQUIVALENT I-131 $\leq$ 60 μCi/gm.	Once per 4 hours
	<u>AND</u>		
	A.2	Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> DOSE EQUIVALENT XE-133 not within limit. <u>OR</u> DOSE EQUIVALENT I-131 > 60 μCi/gm.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.16.1	---------------------------NOTE--------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 500 μCi/gm.	7 days
SR 3.4.16.2	---------------------------NOTE--------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

-----NOTE-----
The control room envelope (CRE) and control building envelope (CBE) boundaries may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, and 4,
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREVS train inoperable for reasons other than Condition B.	A.1	Restore CREVS train to OPERABLE status.	7 days
B. One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary in MODES 1, 2, 3, or 4.	B.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	B.2	Verify mitigating actions to ensure CRE occupant radiological exposures will not exceed limits and CRE occupants are protected from chemical and smoke hazards.	24 hours
	AND		
	B.3	Restore CRE boundary and CBE boundary to OPERABLE status.	90 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	D.1 Place OPERABLE CREVS train in CRVIS mode. OR D.2.1 Suspend CORE ALTERATIONS. AND D.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately
E. Two CREVS trains inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS. OR One or more CREVS trains inoperable due to an inoperable CRE boundary or an inoperable CBE boundary during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	E.1 Suspend CORE ALTERATIONS. AND E.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately

(continued)

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts, or if open, capable of being closed;

b. One door in the emergency air lock closed and one door in the personnel air lock capable of being closed; and

--NOTE--
An emergency personnel escape air lock temporary closure device is an acceptable replacement for an emergency air lock door.
----

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge Isolation valve.

--NOTE--
Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.