WOLF CREEK
NUCLEAR OPERATING CORPORATION

Cleveland Reasoner
Site Vice President

January 15, 2018
WO 18-0004

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

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

2) 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)

Subject: Docket No. 50-482: Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specifications to Transition to Westinghouse Core Design and Safety 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 basis by adopting the Alternative Source Term radiological analysis methodology in accordance with 10 CFR 50.67, "Accident Source Term." Reference 2 provided a request for additional information related to the application. Attachments I through III provide WCNOC's response to the request for additional information, proposed TS changes, and retyped TS pages. Enclosures I and II provide Westinghouse's responses and supplemental responses Attachment I.

Enclosure I provides the non-proprietary Westinghouse Electric Company LLC Attachment 1 to Westinghouse letter SAP-18-2, "Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies." Enclosure II provides the proprietary Westinghouse Electric Company LLC Attachment 2 to SAP-18-2. As Enclosure 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

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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-18-4689, Revision 0, "Application for Withholding Proprietary Information from Public Disclosure," is contained in Enclosure III.

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.

During the review of WCNOC documentation for the Safety Evaluation, the Nuclear Regulatory Commission (NRC) staff noted a discrepancy between Surveillance Requirements (SR) 3.4.1.3 and SR 3.1.4.4. In the March 31, 1999, License Amendment (ADAMS Accession No. ML022050061), the values for Reactor Coolant System (RCS) Flow Rate were the same (37.1 X 10^4, or 371,000). The only "change" requested to SRs 3.4.1.3 and 3.4.1.4 in the April 3, 2001 application for amendment (ADAMS Accession No. ML011000205) was to add the words "and greater than or equal to the limit specified in the Core Operating Limits Report (COLR)" to the minimum RCS flow rates.

In Amendment 144 (ADAMS Accession No. ML020930466) the value for SR 3.4.1.3 was shown incorrectly as 3.71 x 10^4, while SR 3.4.1.4 showed the correct value of 37.1 x 10^4. In spite of this apparent error, when Amendment 144 was incorporated at WCNOC, both SR 3.4.1.3 and SR 3.4.1.4 showed the correct value of 37.1 x 10^4 when released on site. The Surveillance test that satisfies this requirement (STS RE-01, RCS Total Flow Rate Measurement) has always used the TS value of 371,000 (37.1 x 10^4) both before and after the Amendment was incorporated. The incorrect RCS flow rate value was not incorporated at WCNOC in the TS or used in the Surveillances.

As part of the efforts to address the NRC question ARCB1-LLBA-4 provided in Reference 2, regarding Letdown line break analysis (LLBA), the letdown line break dose analysis was revised to model an increased letdown line break flow rate of 222 gpm as well as a reduced water temperature of 290°F and associated airborne fraction. The reduced airborne fraction is included in the response to NRC question ARCB1-LLBA-1. The revised total effective dose equivalent (TEDE) doses are taken from Table 2.0-1 subsequently provided:

Location	Reported Dose (rem)
Exclusion Area Boundary (EAB)	0.20
Low Population Zone (LPZ)	0.065
Control Room (CR)	2.5
Technical Support Center (TSC)	0.25

As part of the efforts to address ARCB1-MSLB-2, the main steamline break dose analysis was revised to model isolation of the main control room concurrent with the end of the faulted steam generator (SG) blowdown at 2 minutes. The revised Main Steam Line Break (MSLB) control room doses are 4.8 rem TEDE for the accident-initiated iodine spike and 4.5 rem TEDE for the pre-accident iodine spike.

As part of the efforts to address ARCB1-CONTROL ROOM-3, the leakage rate into the Refueling Water Storage Tank (RWST) following a Loss of Coolant Accident (LOCA) was

revised to 2.0 gpm. This updated flow rate is reflected in the response to ARCB1-LOCA-4. Additionally, the contribution to the offsite, control room and technical support center doses from the leakage into the RWST was recalculated with the revised leakage rate. The margins reported as part of the response to ARCB1-GENERAL-3 were not affected by the updated calculation.

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) 364-4171, or Cynthia R. Hafenstine at (620) 364-4204.

Sincerely,

Cleveland Reasoner

Cleveland Reasoner

COR/rlt

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

Enclosures: I Attachment 1 to SAP-18-2, "Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies" – Non-Proprietary
II Attachment 2 to SAP-18-2, "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
III CAW-18-4689, 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, w/e (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)

Cleveland Reasoner, of lawful age, being first duly sworn upon oath says that he is Site Vice President 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 ______________
Cleveland Reasoner
Site Vice President

SUBSCRIBED and sworn to before me this 15th day of January, 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

CAW-18-4689, Revision 0, "Application for Withholding Proprietary Information from Public Disclosure"
(9 pages)

Westinghouse

Westinghouse Electric Company
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Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
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CAW-18-4689

January 10, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: SAP-18-2, P-Attachment, "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)

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-18-4689 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-18-4689, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 259, Cranberry Township, Pennsylvania 16066.

JA Gresham

James A. Gresham, Manager
Regulatory Compliance

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, 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: 1/10/18

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, 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-18-2, P-Attachment, "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) 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.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Wolf Creek Nuclear Generation Station

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. "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)

2. "Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies" (Non-Proprietary)

Also enclosed are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-18-4689, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 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 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-18-4689 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2, Suite 259, Cranberry Township, Pennsylvania 16066.

Response to Request for Additional Information

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

Radiation Protection & Consequence Branch (ARCB)

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-LOAC-2 – *LOAC*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-LLBA-1 *(Letdown Line Break)*

1. *Please explain how the flashing fraction was determined. Was the flashing fraction determined consistent with RG 1.183, Appendix A, Regulatory Position 5.4 (which uses a constant enthalpy, h, process based on the maximum time-dependent temperature of the water circulating outside the containment:* $FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$ *where* h_{f1} *is the enthalpy of liquid at system design temperature and pressure;* h_{f2} *is the enthalpy of liquid at saturation condition; and* h_{fg} *is the heat of vaporization at 212 degrees Fahrenheit).*

Response: Regarding the letdown line design system conditions for the analysis documented in Section 4.3.7 of Enclosure IV, the enthalpy of the water was based upon a pressure of 600 psig and a temperature of 380°F. As stated in Section 15.6.2 of the USAR, the letdown orifices reduce the letdown line pressure from 2,235 psig to less than 600 psig outside containment during normal plant operation and thus 600 psig is an appropriate value to use. For the temperature value, as shown in Table 9.3-9 of the USAR, the design outlet temperature of the regenerative heat exchanger (upstream of the letdown orifices) is 290°F. While 290°F is the design temperature of the system, the value had been increased well above the design temperature to 380°F for the analysis.

However, while developing the responses to this question and ARCB1-LLB1-4, it was determined that it is more appropriate to use a temperature of 290°F as it is consistent with the CLB (the CLB value of 286°F has been rounded up) and the design conditions of the system (Table 9.3-9 of the USAR). Therefore, the temperature value used for calculating enthalpy has been changed to 290°F.

Since the enthalpy value has been changed as part of this response, the flashing fraction has also been recalculated based upon the updated enthalpy value.

A supplemental response to this request is provided by Westinghouse in Enclosure I.

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

<u>RAI ARCB1-LLBA-3</u> *(Letdown Line Break)*

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

<u>Response:</u>

The auxiliary building boundary is credited in the Letdown Line Break (LLB) analysis and other design basis radiological consequences analyses that involve a release to the auxiliary building (e.g., LOCA).

The auxiliary building emergency HVAC equipment, Emergency Exhaust System (EES), is not credited in the LLB analysis. However, it is credited within the LOCA doses analysis for a release to the auxiliary building.

The auxiliary building normal HVAC equipment is not credited in the LLB analysis or any other design basis radiological consequences analysis.

The auxiliary building boundary complies with Regulatory Guide 1.183 Position 5.1.2 as it is safety-related and required to be operable by technical specifications (during the modes of interest). As the building boundary does not require power or automatic actuation to perform its safety function, these two requirements of Position 5.1.2 are not applicable.

Regarding the EES, for the LOCA doses analysis this system complies with Regulatory Guide 1.183 Position 5.1.2 as it is safety-related, required to be operable by technical specifications (during the modes of interest), powered by emergency power sources, and automatically actuated (Safety Injection Signal). However for the LLB analysis, as the EES many not be automatically actuated, it is credited for mitigation of the event.

Note that while the auxiliary building normal HVAC equipment is not credited to mitigate the event, it is modeled to produce a more limiting event. Specifically, it is recognized that it will take time for the operators to identify that a letdown line break has occurred and respond accordingly (analysis assumes 30 minutes). Thus, prior to detecting the letdown line break, the non-safety related auxiliary building normal exhaust may be in operation rather than the safety

related EES. Since the auxiliary building normal exhaust utilizes non-safety related filters, no credit for filtration is taken.

Regarding the ability of the auxiliary building normal exhaust to transport radionuclides released within the auxiliary building to the unit vent, per Section 9.4.3.1.2 of the Updated Safety Analysis Report (USAR), the Power Generation Design Basis Three for the auxiliary building normal exhaust is:

> POWER GENERATION DESIGN BASIS THREE - The auxiliary/fuel building normal exhaust system exhausts slightly more air than is being supplied, to inhibit exfiltration of the air from the auxiliary building. The main steam tunnel exhaust system exhausts an amount of air equal to that being supplied.

Therefore, while no credit is taken for filtration of the release, if the auxiliary building normal exhaust is in operation, it will be able to transport radionuclides to the unit vent.

While either the auxiliary building normal exhaust or the EES should be in-service during a letdown line break, Section 9.4.3.2.1.a of the USAR discusses the scenario where no auxiliary building exhaust systems are in service (emphasis added):

> The auxiliary/fuel building normal exhaust system is provided with redundant, full-capacity fans. However, **assuming a loss of the exhaust air flow, the supply system automatically shuts down to prevent building pressurization**.
>
> Therefore, a postulated loss of the exhaust system results in a complete loss of direct outside air movement within the auxiliary building. Natural air flow patterns may be established, depending on thermal gradients and the flow paths existing within and across the auxiliary building. Assuming uniform mixing of the auxiliary building atmosphere as the most conservative case, **there would be negligible effect in relation to operator exposure if the ventilation system is returned to service within several hours**. Actions are taken to remove unnecessary equipment from service if it contributes to personnel exposure in order to maintain exposures ALARA.
>
> The loss of normal ventilation will have no impact on those areas with safety-related equipment. Other areas of the building are periodically monitored, depending upon operating loads and duration of the loss of ventilation, to determine the impact on equipment.

Regarding a letdown line break, it is assumed to occur upstream of the letdown heat exchanger, as the outlet of the letdown heat exchanger will be well below 212°F (and thus preclude flashing) in order to support the maximum allowable resin bed operating temperature (approximately 140°F, as stated in Section 9.3.4.2.2 of the USAR). The letdown heat exchanger is located in the basement of the auxiliary building (1974' elevation). The entrance to the control room envelope is at the 2047' elevation (5 floors above the 1974' elevation). Due to the torturous path from the auxiliary building basement to the control room envelope, the dilution that would occur as the steam traveled through the auxiliary building, and due to the fact that both building boundaries will be required to be operable by TS, the amount of radioactivity that would travel to the control room envelope with no HVAC equipment in service is judged to be negligible, consistent with Section 9.4.3.2.1.a of the USAR. On the other hand, if the auxiliary building normal exhaust is in operation, 100% of the flashed steam is released directly to the unit vent with no credit for holdup, dilution, or filtration. Therefore, while the auxiliary building normal exhaust system is not safety related, it results in a more limiting event by

instantaneously releasing the flashed steam to the environment rather than allowing it to remain within the auxiliary building.

If no exhaust system is in operation during a Letdown Line Break, the release will remain in the auxiliary building for the first 30 minutes of the event. Once the operators have identified and isolated the letdown line break, EES and its associated filters could be used to limit the release from the auxiliary building to the environment; however, in order to model the limiting scenario of normal exhaust operating throughout the event, a direct release to the environment with no credit for holdup, dilution, or filtration is modeled for the duration of the event.

In summary, the auxiliary building normal exhaust is not credited in the Letdown Line Break analysis. Rather, it is modeled to yield a more limiting event as it results in a direct release to the environment with no credit for holdup, dilution, or filtration rather than allowing the release to remain within the basement of the auxiliary building.

A supplemental response to this request is provided by Westinghouse in Enclosure I.

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

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

Response: The 141 gpm value corresponds to the maximum break flow corresponding to the limiting letdown lineup procedurally allowed during normal operation. As stated in Section 9.3.4.2.3.2 of the USAR (emphasis added):

> During normal operation, the letdown flow is 75 or 120 gpm, and one mixed bed demineralizer is in service. Reactor coolant samples are taken at frequent intervals to check boron concentration, water quality, pH, and activity level. The normal charging-pump flow control valve (FCV-462) is modulated by the pressurizer water level at the set point programmed for a prevailing reactor coolant average temperature. **During normal operation with maximum purification, the letdown flow is 120 gpm.** If a standby centrifugal charging pump is employed, the charging flow control valve (FCV-121) is modulated by pressurizer water level.

As such, a letdown line break flow of 141 gpm is expected to bound all letdown line breaks that could occur during normal operation.

The current licensing basis analysis is based upon a letdown line break flow of 222 gpm. The 444 gpm value is based upon doubling the break flow of 222 gpm, as discussed in Section 15.6.2 of the USAR. The 222 gpm break flow corresponds to a letdown lineup that is not procedurally allowed during normal operation, i.e., it assumes all letdown lines are in use prior to the break occurring.

As previously stated, the 141 gpm value is expected to bound the limiting letdown line break flow that could occur during normal operation. Nevertheless, in order to ensure a bounding Letdown Line Break analysis, the 141 gpm value used for the Alternative Source Term (AST) Letdown Line Break analysis will be increased to 222 gpm to be consistent with the CLB.

However, the AST analysis does not double the break flow rate value as the activity is only being released from the end of the break connected to the RCS. Thus, it is not necessary to double the break flow for this analysis, consistent with standard Westinghouse analysis approach.

In addition to this change, as documented within the response to ARCB1-LLBA-1, the flashing fraction was also recalculated based upon a more accurate value of 290°F.

Based upon these two changes, the letdown line break analysis was revised. The resulting letdown line break accident doses are listed below:

- EAB 0.20 rem TEDE
- LPZ 0.065 rem TEDE
- Control room 2.5 rem TEDE
- TSC 0.25 rem TEDE

The EAB dose reported is for the worst 2 hour interval, determined to be from 0 to 2 hours.

<u>RAI ARCB1-LOCA-1</u> *- Loss-of-Coolant Accident (LOCA)*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

<u>RAI ARCB1-LOCA-2</u> *- LOCA Control Room Modeling*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

<u>RAI ARCB1-LOCA-3</u> *- LOCA and other DBAs except Fuel Handling Accident (FHA)*

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

   or

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

Response: The proposed change for this request is provided in Attachments II and III. Attachment II and III provide the Proposed Technical Specification Changes (Mark-up) and Revised Technical Specification Page, respectively.

RAI ARCB1-LOCA-4 *- LOCA*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-LOCA-5 *– 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 OBA and describe how these assumptions are considered in the radiological analyses. It does not appear to be realistic to assume the containment is isolated at the beginning of the event unless the containment is not allowed to be unisolated during operations. If this is the case please state so, and justify this answer.*

Response: Per Section B 3.6.3 of the TS Bases (emphasis added):

> The ACTIONS are modified by a Note allowing penetration flow paths, except for 36 inch containment purge supply and exhaust valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the **penetration can be rapidly isolated when a need for containment isolation is indicated**. Due to the size of the containment purge line penetration and the fact that those **penetrations exhaust directly from the containment atmosphere to the environment via the unit vent**, the penetration flow path containing these valves may not be opened under administrative controls. A single valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

Thus, consistent with the Standard Technical Specifications (STS) bases, any penetration flow path that is unisolated must be done so under administrative controls that ensure that the containment penetrations can be closed consistent with the safety analysis. As such, since the containment purge line is a direct path between the containment atmosphere and the environment, this penetration is not allowed to be unisolated due to the possibility of bypassing the containment barrier.

As stated in the response to ARCB-RAI-38, the basis for deleting the wording from the Bases for TS 3.6.3 is that it refers to the containment purge and exhaust valves and not the mini-purge valves. Regarding the mini-purge values, they are isolated within 10 seconds of accident initiation as explicitly stated in the marked up TS Based provided within Enclosure IV.

Therefore, the assumed time for the containment to isolate after initiation of the applicable design basis accidents (LOCA and rod ejection) is 10 seconds.

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

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

   or

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

Response: The proposed change for this request is provided in Attachments II and III. Attachment II and III provide the Proposed Technical Specification Changes (Mark-up) and Revised Technical Specification Page, respectively.

RAI ARCB1-FHA-2 *- FHA*

The NRC question and Westinghouse response to this request is provided in Enclosures I and II.

RAI ARCB1-FHA-3 *- FHA*

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

Response: The AST Fuel Handling Accident (FHA) accident analysis assumes the movement of an irradiated fuel assembly at the earliest permitted time of greater than or equal to 72 hours from removal of the critical reactor core. The FHA in Containment assumption of the failure of 120% of the rods of one entire irradiated fuel assembly and the FHA in the Fuel Building assumption of the failure of 100% of the rods of one entire irradiated fuel assembly are consistent with the current licensing basis limiting cases. The use of the AST methodology does not result in a change in these limiting cases.

The movement of loads other than irradiated fuel, including control rods, sources, and fresh fuel, is administratively controlled to ensure that the equipment used to handle loads within the reactor cavity and SFP regions will function as designed and that the equipment has sufficient load capacity for handling loads. These administrative controls used to be contained in the TS as TS 3/4.9.6, "Refueling Operations - Refueling Machine," TS 3/4.9.7, "Refueling Operations: Crane Travel - Spent Fuel Storage Facility," and TS 3/4.9.10.2, "Refueling Operations: Water Level – Reactor Vessel;" however, the NRC previously approved relocation of these administrative controls from the TS to the WCNOC USAR (Chapter 9.1) as part of NRC License Amendment 89 dated October 2, 1995. The NRC approval was based on the administrative requirements not falling within the criteria for mandatory inclusion in the TS in 10 CFR 50.36.

RAI ARCB1-FHA-4 - *FHA*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-FHA-5 - *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 percent of the free volume) and the fuel building on a case-by-case basis, but no guidance exists for mixing in the auxiliary building.*

Response: It is worth noting that the analysis was performed in response to a specific acceptance review question. As the FHA is typically analyzed as a direct release to the environment rather than a release within the neighboring building (for WCGS the auxiliary building), no generic methodology was available when developing the analysis. As such, it was necessary to develop a conservative method to analyze the scenario. Additionally, judgement was used to select conservative inputs and assumptions for the accident.

Regarding the limiting single failure considered for this event, it is the failure of a containment gaseous radiation monitor or its associated power supply. As shown in TS Table 3.3.6-1, one channel of the containment atmosphere gaseous radiation monitor must be operable during the movement of irradiated fuel assemblies. If this monitor does not fail, then it will actuate a containment purge isolation signal, which will in turn actuate a Control Room Ventilation Isolation Signal (CRVIS). Once a CRVIS is generated, the control room will be pressurized and the primary dose contributor, inleakage from the auxiliary building, will be terminated and the event will be non-limiting.

In order to aid the response to this question (as the accident is not typically analyzed in this manner), two figures have been enclosed with this Attachment (Enclosure I and Enclosure II). The first attachment shows a marked up floor plan of the relevant auxiliary building level. The second attachment shows all containment penetration locations.

In regards to "stack effects," as shown on Enclosure I and Enclosure II, the only two penetrations that would allow direct movement of air between the containment building and the outside environment are the equipment hatch and the auxiliary access hatch (also referred to as the emergency air lock). While the equipment hatch could be open following a FHA, as stated in the response documented within ET 17-0011 the emergency air lock will not be:

> *Regarding the emergency air lock, this pathway is isolated during fuel movement and will not serve as a release point following a FHA. Specifically, the note associated with LCO 3.9.4.b regarding temporary closure devices was approved per the Safety Evaluation for Amendment Number 74 (Accession Number ML02204037). As stated in the Safety Evaluation, "The staff finds that the proposed change provides an effective means of preventing the release of radioactive material following a fuel handling accident." Thus, a release from the emergency air lock is not analyzed for the FHA.*

Thus, as "stack effects" are induced by movement of air into and out of buildings, and since only one penetration could be open that would allow movement of outside air into or out of the containment, WCGS does not consider "stack effects" as a credible source of pressurization following a FHA.

As for pressurization due to winds, the analysis and associated inputs and assumptions documented within ET 17-0011 are valid for wind speeds expected during a FHA. This is based upon a number of factors. First, in order to quantify the wind pressurization, a "simplified" version of the methodology outlined in ASCE 07-05 was utilized. In regards to "simplified," this refers to the fact that the coefficients that are included in the base equation which consider various factors (such as nearby topography), that serve to reduce the overall wind pressurization, were all set equal to 1.0 in order to maximize the resulting pressure increase. The resulting simplified equation is given below:

$$P = 0.00256 * V^2$$

Where:

P	=	wind pressurization (psf)
V	=	wind speed (mph)

As documented in Table 4.1.1-19 of Enclosure IV, only 2.08% of the wind speed frequency was greater than 10.00 m/s (22.4 mph). As previously stated, since no generic guidance exists for analyzing this event, WCGS applied methods for similar scenarios to this case. Specifically, Regulatory Guide 1.194 documents that a 95% χ/Q value should be utilized for dose analyses. As meteorological data are the primary contributor for the 95% χ/Q, it was judged that it is reasonable to use a 95% bounding wind speed (for this case, the maximum) for the FHA analysis. However, while 2.08% of the time the wind speed exceeded 10 m/s (22.4 mph), this frequency does not take into account that the wind would need to be from a specific sector in order to reach the maximum pressurization.

Regarding wind direction, the equipment hatch is 128° clockwise from the north direction (7° from the south east sector). Thus, only the E, ESE, SE, SSE, and S wind directions will be able to pressurize the containment via the equipment hatch.[1] Based upon the meteorological data in

[1] While it is recognized that a wind from the ENE direction could enter the equipment hatch, the resulting angle difference (67.5° compared to the equipment hatch of 128°) would result in a greater than 50% reduction in magnitude (cos(60.5°) = 0.49).

Table 4.1.1-1 through Table 4.1.1-7, the wind speed exceeds 8 m/s for these 5 sectors less than 4.0% of the time. Thus, if wind direction is considered, the wind speed will be less than 8.0 m/s from a wind direction that could pressurize containment greater than 95% of the time. Thus, it is reasonable to use 8.0 m/s (17.9 mph) for the wind pressurization calculation.

If a wind speed of 8.0 m/s (17.9 mph) is substituted into the wind pressurization calculation, the resulting pressure increase is less than 0.16 inches of water gauge (in w.g.). Furthermore, as this calculation does not credit the reduction of wind speed for an angled entry, the wind speed (and resulting pressurization increase) could further be reduced if the true 95% wind speed value was calculated. However, since acceptable results were obtained with a bounding wind speed, it was not necessary to calculate the 95% wind speed value for the equipment hatch.

The analysis conservatively assumes that the wind pressurization is instantaneously translated to within containment as well as the neighboring auxiliary building. This is a conservative assumption as it takes no credit for a pressure drop due to the wind passing through the equipment hatch, or for any pathway that air would take from within the containment to the auxiliary building. Additionally, while incoming air would dilute the source in containment, the analysis does not credit any dilution due to air being blown into containment. Moreover, this value assumes that the wind pressurization is constant throughout the duration of the event (30 minutes). If the wind changed direction, or decreased, the wind pressurization would decrease and air would exit containment via the equipment hatch. Based upon these conservative assumptions and the lack of generic guidance for this event, it was judged that considering a wind pressurization of less than 0.16 in w.g. is appropriate.

Next, regarding the flows assumed from the containment to the auxiliary building, two scenarios were considered. The first scenario is that the auxiliary building normal exhaust is running at the onset of the event. However, as the auxiliary building normal exhaust system is not safety related, an additional scenario where no exhaust fans are running was considered to demonstrate that the auxiliary building normal exhaust is not credited to mitigate the event; rather, it is explicitly modeled to make the event worse.

In order to develop a bounding value (a larger flow rate will result in a faster buildup of activity in the auxiliary building and in turn operator dose) for the normal exhaust flow rate, the design value for flow rate from the 2047' elevation of the auxiliary building (intake locations are shown on Enclosure I) of 3,250 cfm was used as a starting point. The HVAC lineup is procedurally tested to ensure that flow rates are within ±10% of design conditions. In order to bound this, the design value of 3,250 cfm was increased by 50% to a value of 4,875 cfm. As this value is for non-safety related equipment that is modeled to make the event more limiting, it was judged that the value was bounding for this application.

Once the control room HVAC system is placed in the emergency lineup, the pressurization of the Control Room Envelope (CRE) will terminate inleakage from the auxiliary building. Thus, in order to maximize the dose, the emergency exhaust flow rate is maximized in order to increase the amount of radioactivity transported from within the auxiliary building to the environment. In order to bound this value, the design value of 2,225 cfm was doubled (in case 2 trains are placed into operation) and then increased by 50% to bound actual fan performance compared to design conditions (total of 6,675 cfm). Due to the conservatism applied to this parameter, it was judged to be a bounding input.

Regarding the flow rates from the containment to the auxiliary building, since no generic guidance is available for the event, engineering fundamentals were applied to conservatively model the event. For this specific input (flow rates), conservation of mass was utilized. Thus,

the mass of air that moves into the auxiliary building is equal to the mass of air exiting the auxiliary building. The amount of air entering the auxiliary building is equal to the mass of air leaking into the control room envelope plus the mass of air being removed from the auxiliary building via the exhaust system applicable for the condition analyzed (normal exhaust, EES, or no exhaust).

For the auxiliary building volume credited, Enclosure I shows which volumes were and were not considered. The open area between the containment and the CRE envelope is shown in blue and was credited as a mixing volume. As shown on the drawing, for simplicity and conservatism, the air volume directly next to the containment wall arc was neglected (shown in white). As the roof for this floor is 24 feet, the approximate volume of this area is greater than 178,000 ft^3. This volume was reduced to 70,000 ft^3 to account for equipment as well as the percentage available for mixing.

The 70,000 ft^3 volume also includes additional conservatisms. First, the volume shown in orange would likely be available for dilution, but it is conservatively neglected. As shown on Enclosure I and Enclosure II, only three penetrations are at the 2047' or above elevation and are connected to the volume marked in blue on Enclosure I.

The first penetration is for the purge line. Per TS LCO 3.3.6 Condition B, if the containment purge isolation system has one or more functions with one or more channel inoperable, the penetration must be closed during movement of irradiated fuel assemblies. If all functions are available for the containment purge isolation and if the penetration is in use, a containment purge isolation signal will be generated which will both automatically isolate the purge line and initiate a CRVIS which will terminate the unfiltered inleakage into the CRE. Thus, the containment purge penetration is not a credible release path for this event.

The second penetration is a spare and is blanked off and therefore is not a credible release path either.

The third penetration is for the H^2 analyzer line. This line is a closed system that contains radiation monitors with the capability to generate a containment purge isolation signal as well as a CRVIS. Thus, unless major maintenance was being performed on the line at the time of the FHA, it would be expected that the line would either be isolated to the auxiliary building or the penetrations would be open to a closed line that would be capable of generating a CRVIS. As stated in TS LCO 3.9.4, any unisolated containment penetrations need to be done so in accordance with administrative controls. The intent of the administrative controls is to ensure that if the penetration needs to be isolated following a FHA, that it can be done so rapidly. Thus, major maintenance (i.e., opening the system up to the environment which could not quickly be isolated) on the H^2 analyzer line during movement of irradiated fuel would not be performed. Additionally, as previously stated, since the single failure for the scenario is the failure of a containment gaseous radiation monitor or its associated power supply, there is not a credible failure that would allow containment atmosphere to pass through the H^2 analyzer line into the auxiliary building.

As WCGS does not consider a release through the three containment penetrations located on the 2047' elevation as credible following a FHA, it would be reasonable to include the volume marked in orange on the 2047' elevation as any radiation released through the containment personnel hatch would have this volume of air available for dilution. Furthermore, as a wall separates the orange and blue volumes broken only by a fire door and transfer grill, the orange area would need to be pressurized first before flowing into the blue volume. Nevertheless, in

order to use a bounding and simplified dilution volume, the orange volume in Enclosure I is not credited as part of the 70,000 ft^3 value.

Moreover, it is worth noting that for any penetrations below the 2047' elevation, the path from the associated penetration to the CRE is more torturous than the path considered within this analysis. As such, a release from any other penetration (other than the equipment hatch which is analyzed as the licensing basis event) is bounded by this analysis. As the analysis did not credit a pressure drop across a penetration opening and no flow reduction is credited due to penetration size, the analysis is independent of the penetration opening area (both for the individual penetration as well as the total).

In addition to not crediting the orange volume, the green volume (CRE) in Enclosure I is also not credited. If the volume was credited, then once the unfiltered inleakage entered the CRE, the volume would be available for further dilution prior to leaking into the HVAC equipment. However, in order to use a bounding and simplified dilution volume, the green volume is not credited as part of the 70,000 ft^3 value.

Based upon the lack of generic guidance for calculating a value for this scenario, and due to the conservatisms in the 70,000 ft^3 volume credited for dilution, it is judged that this is a reasonable value to use within the analysis.

Regarding the unfiltered inleakage into the CRE value of 60 cfm, this value is based upon crediting two groups of sealed barriers in series. First, as stated in the justification of the 70,000 ft^3 volume, WCGS does not consider that there is a credible path from the three containment penetrations located on the 2047' elevation to the auxiliary building open area. As such, the credible path is through the containment personnel hatch. As shown in Enclosure I, there is a wall penetrated only by a transfer grill and fire door between the area directly outside the personnel hatch and the main open area (shown in blue) on the 2047' elevation of the auxiliary building. While this door is expected to be closed during an event as it is credited as a fire barrier and the transfer grill is a relatively small area, the presence of this barrier is conservatively neglected within the analysis. Thus, the remaining two sets of barriers in series are the 1) three airtight doors in parallel and 2) two closed isolation dampers on the control room HVAC intake duct.

As previously stated, the pressurization in containment due to wind was calculated to be less than 0.16 in w.g. The two isolation dampers (one per each train) are rated at less than 30 cfm leakage apiece at 6 in w.g. For conservatism, the analysis modeled 60 cfm (30 cfm x 2). As the wind pressurization was calculated to be less than 0.16 in w.g., this is a bounding value. Additionally, the presence of three airtight doors would serve to further reduce the inleakage value. Nevertheless, the 60 cfm value is utilized to ensure the scenario is bounding.

The 60 cfm unfiltered inleakage into the control room was applied to both the auxiliary building normal exhaust fan running scenario, as well as the scenario where no exhaust is running. For the limiting of the two scenarios (auxiliary building normal exhaust running), the normal exhaust will create a vacuum that will offset the wind pressurization. Specifically, per SR 3.7.13.4, the EES is tested to ensure that it can create a negative pressure of greater than 0.25 in w.g., which is greater than the wind pressurization value. As stated in the discussion about the auxiliary building exhaust flowrate, the auxiliary building normal exhaust system is designed to pull more air from the 2047' elevation of the auxiliary building (3,250 cfm versus 2,225 cfm). Thus, if the auxiliary building normal exhaust is in service, it is expected that the normal exhaust would offset the pressurization due to wind and thus prevent inleakage into the CRE. Nevertheless, in order to bound the scenario, 60 cfm was utilized for both cases.

For the case where no exhaust is modeled, due to the non-limiting nature of the scenario (dose was calculated to be 0.012 rem TEDE), in order for the scenario to become limiting, the input parameters (specifically the dilution volume and/or the unfiltered inleakage) would need to be changed substantially (dilution volume would need to be decreased and the unfiltered inleakage would need to increase). That being said, as justification has been provided for each of the input parameters, WCGS has judged that all values are appropriate for use.

In summary, due to a lack of generic guidance, significant margin was applied to the input parameters for use in the analysis in order to demonstrate the nonlimiting nature of the event. Due to large conservatism applied to the various parameters, WCGS considers this a non-limiting event that is bounded by the licensing basis FHA (release through an open equipment hatch). If realistic inputs were utilized, the resulting dose of a release through an open personnel hatch would be reduced even further.

Reference to ASCE 07-05

Mehta, Kishor C, "Wind Loads: Guide to the Wind Load Provisions of ASCE 07-05," 2010 by the American Society of Civil Engineers.

RAI ARCB1-FHA-6 - *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 Appendix A to 10 CFR Part 50 and 10 CFR 50.67. In addition, please provide the inputs, assumptions, methodology a technical basis for the analysis, and justify the assumptions used.*

Response: The various containment penetrations, and various combinations of multiple penetrations, are addressed by the licensing basis event described in Section 4.3.12 of Enclosure IV and the analysis documented within ET 17-0011. The justification for how each penetration (and combination of multiple penetrations) is addressed is as follows:

Equipment Hatch

This penetration is explicitly addressed by the licensing basis event described in Section 4.3.12 of Enclosure IV.

Personnel Air Lock

This penetration is addressed by the analysis documented within ET 17-0011.

Specifically, the analysis allows for a containment to auxiliary building penetration to be open for the duration of the event. Additionally, as discussed in the response to ARCB1-FHA-5, the analysis inputs and assumptions allow for the equipment hatch to be open for the duration of the event (which could result in pressurization of the containment due to wind). As the analysis does not credit any radiation leaving through the equipment hatch, it bounds the scenario where the equipment hatch is closed while the personnel air lock is open.

Emergency Air Lock

As documented within the response within ET 17-0011, the emergency air lock will not be open during a FHA. The following justification was provided:

> *Regarding the emergency air lock, this pathway is isolated during fuel movement and will not serve as a release point following a FHA. Specifically, the note associated with LCO 3.9.4.b regarding temporary closure devices was approved per the Safety Evaluation for Amendment Number 74 (Accession Number ML02204037). As stated in the Safety Evaluation, "The staff finds that the proposed change provides an effective means of preventing the release of radioactive material following a fuel handling accident." Thus, a release from the emergency air lock is not analyzed for the FHA.*

Containment Penetrations above the 2047' Elevation

As shown on Enclosure I and Enclosure II to this attachment, there are only 6 penetrations (besides the personnel air lock and the equipment hatch) above the 2047' Elevation. These six penetrations are for the purge lines (two penetrations), H^2 analyzer lines (two penetrations), and two blanked off spare locations. As described in the response to ARCB1-FHA-5, justification exists for crediting these penetrations as isolated during a FHA. Nevertheless, in order to conservatively account for these penetrations, a reduced mixing volume within the auxiliary building was modeled. As such, these penetrations are addressed by the analysis documented within ET 17-0011.

Containment Penetrations below the 2047' Elevation

For the remaining penetrations, as shown on Enclosure II to this attachment, they are below the 2047' elevation. Besides one penetration, the fuel transfer tube, all penetrations are connected to the auxiliary building. Thus, any release would be to a lower level of the auxiliary building than the 2047' floor (floor that contains the entrance to the CRE). The resulting release to a lower level would have a more torturous path than a release on the 2047' level and be further diluted. Thus, these penetrations are bounded by the analysis within ET 17-0011.

For the fuel transfer tube, as it is connected to the fuel building, any leakage through it would be bounded by a fuel handing accident within the fuel building.

Combination of Penetrations

The analysis documented within ET 17-0011 accounts for an open equipment hatch and open penetrations within the auxiliary building. Specifically, the flow rates bound the flow rates expected due to pressurization of the containment due to wind. Additionally, the analysis used conservation of mass to determine bounding flow rates (i.e., mass flow rate in equals mass flow rate out). The analysis did not credit a pressure drop due to air passing through penetrations and it did not credit a corresponding reduction in flow rate. As such, the analysis is not limited by the number or size of penetrations open. In essences, whatever air mass leaves the auxiliary building (via the auxiliary building exhaust and leakage into the control room) is instantly made up by incoming air from the containment atmosphere, with no credit for how the air has to move to get there. As such, the analysis documented within ET 17-0011 is independent of the number and size of penetrations open to the auxiliary building.

Summary of Results

The licensing basis event described in Section 4.3.12 of Enclosure IV addresses an open equipment hatch. The analysis documented within ET 17-0011 addresses all other containment penetrations (including combination of various penetrations). The resulting doses are subsequently shown (Note that the analysis within ET 17-0011 considered two cases: auxiliary building normal exhaust in-service and no exhaust in-service):

	Control Room Dose from Equipment Hatch [Section 4.3.12 of Enclosure IV]	Control Room Dose from Other Containment Penetrations[1] (normal exhaust in-service) [ET 17-011]	Control Room Dose from Other Containment Penetrations[1] (normal exhaust off) [ET 17-011]
Dose =	0.92 rem TEDE	0.77 rem TEDE	0.012 rem TEDE
Limit =	5 rem TEDE	5 rem TEDE	5 rem TEDE

1 Addresses all combination of open containment penetrations other than a release through the equipment hatch to the environment

Inputs, Methods, and Assumptions

The inputs, methods, and assumptions for the licensing basis FHA are provided within Section 4.3.12 of Enclosure IV. Additionally, as documented within the response to ARCB1-GENERAL-3, 10% margin was applied to the resulting control room doses.

Regarding the analysis developed to address the remaining containment penetrations, the inputs, methods, and assumptions were provided within ET 17-0011 and additional justification is provided in the response to ARCB1-FHA-5. Specifically, the following values were modeled:

A containment volume of 1.25E+06 ft^3 is modeled as it is equal to 50% of the minimum free volume.

A conservative auxiliary building volume of 7.0E+04 ft^3 is modeled based on the volume of the rooms located between the containment penetration and the control room (additional details for the value, and justification for its use, are provided within the response to ARCB1-FHA-5).

The analysis considers cases with no exhaust (0 cfm), normal exhaust (4875 cfm), and emergency exhaust (6675 cfm). Additional details for the exhaust flow rates, and justification for their use, are provided within the response to ARCB1-FHA-5. No credit for emergency exhaust filtration was taken for the flow leaving the auxiliary building.

The flow rate from containment to the auxiliary building is based upon conservation of mass principles. Specifically, the mass flow rate into the auxiliary building is set equal to the mass flow rate leaving the auxiliary building (applicable exhaust flow rate plus leakage into the control room).

Leakage into the control room from the auxiliary building is 60 cfm based on a conservative estimate of leakage through the airtight control room envelope door and leakage from the operating train of HVAC isolation damper. Additional details on this parameter, and justification for its use, are provided within the response to ARCB1-FHA-5.

Switchover from normal mode HVAC operation to emergency mode HVAC operation in both the auxiliary building and control room is assumed at 30 minutes after event initiation. This 30-

minute time is significantly longer than the generally accepted time of 10 minutes, consistent with Section 3.1.2.d of the Wolf Creek USAR, to identify an event and manually switch to emergency mode. In emergency mode, the control room is pressurized such that inleakage from the auxiliary building to the control room is terminated.

These inputs were incorporated into to the base case documented within Section 4.3.12 of Enclosure IV and then analyzed by using the RADTRAD computer code.

RAI ARCB1-FHA-7- *FHA*

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

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

Response:

1. For the control room doses, the worst case single failure is the failure of one of the filtration fans at the start of the emergency mode of operation. For the offsite doses, no explicit single failure is modeled as there is not a failure that will result in a more limiting analysis.

2. The worst case single failure for the control room doses and offsite doses is stated in the response to part 1 of this question. The justification for the failures are subsequently provided.

   The current licensing basis analysis accounts for a failure of the humidity controller in one train of EES (described in Section 15.7.4.5.1.2.k. of the USAR) which results in the filter efficiency decreasing to 82.5%. For the FHA analysis documented in Enclosure IV, no credit is taken for EES filtration. As such, regardless of if the humidity controller failure is considered or not, the EES filter efficiency credited in the analysis will be 0% (i.e., the failure of a humidity controller has no impact on the results of the analysis).

   As stated in TS LCO 3.7.13, two EES trains shall be operable during movement of irradiated fuel assemblies in the fuel building. Thus, if a single failure of one train of EES is considered, the remaining second train of EES will be able to perform its associated safety function of transporting the radioactivity released within the fuel building to the unit vent.

   In summary, there is not a single failure that will result in either a decreased EES filtration credited within the analysis or a change to the release point utilized for dispersion factors. Thus, the failure of one of the control room filtration fans at the start of the emergency mode of operation is the limiting single failure for control room doses. Due to the conservative modeling for the FHA and the equipment required to be operable by TSs, there is not a single failure that will result in more limiting offsite doses.

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-SGTR-2 *- SGTR*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-SGTR-3*- SGTR*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-SGTR-4*- SGTR*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-SGTR-5*- SGTR, 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 Appendix A to 10 CFR Part 50 and 10 CFR 50.67. Note this case would be consistent with the proposed and current TS Bases, which states that: In both analyzed cases for the noble gas specific activity is assumed to be equal to or greater than 500 μCi/gm DOSE EQUIVALENT XE-133. In addition, provide the inputs, assumptions, methodology technical basis for the analysis and the results of the analysis. Also, please justify the assumptions and inputs used in the analysis.*

   or

2. *Please provide a proposed change to TS 3.4.16 that is consistent with the analyses proposed in the LAR. Note that an example of what has been found acceptable to the NRC staff can be seen with the treatment of Dose Equivalent 1-131 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).*

Response: The dose analyses model noble gases initially in the primary coolant at a concentration of 500 μCi/gm dose equivalent Xe-133. This represents an upper limit on allowable noble gas coolant concentration. The Technical Specification Bases for LCO 3.4.16 will be revised to remove the words "or greater than." The revised sentence is:

"In both cases, the noble gas specific activity is assumed to be equal to 500 μCi/gm [micro-Curies per gram] DOSE EQUIVALENT XE-133."

As discussed in the letter referenced within ARCB1-SGTR-5 (Adams Accession No. ML16113A402), this question is associated with a generic issue with Technical Specification Task Force (TSTF)-490, Revision 0. As such, the Technical Specifications Task Force is developing a response to address the generic issue. As WCGS had previously implemented this TSTF under License Amendment 170 (ADAMS Accession No. ML062790364), this submittal is not requesting a change to the current completion time of 48 hours.

In regards to the letter to the Technical Specification Task Force from the NRC, the following is stated (emphasis added):

> **Typically, the Required Action for a condition not analyzed requires the plant to take immediate actions to begin shutdown of the plant.** This action is consistent with the current required action for exceeding the gross specific activity of the reactor coolant, which requires the plant be in Mode 3 with Tavg less than 500°F within 6 hours.

While the example provided (exceeding gross specific activity) does support the statement that a condition not analyzed requires immediate actions to begin shutdown of the plant, there are several examples that support the opposite position that a plant is allowed a reasonable amount of time to return the parameters of interest to within associated limits. For example, the following LCOs within Standard Technical Specifications allow for time to return the parameters of interest to within limits:

> LCO 3.5.4 allows for 8 hours to restore the Refueling Water Storage Tank (RWST) to operable status if the boron concentration or temperature is not within the associated limits
>
> LCO 3.6.5 allows for 8 hours to restore the containment average air temperature to within limits if the temperature exceeds associated limits
>
> LCO 3.7.10 allows for 24 hours to verify mitigating actions are implemented that ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits due to an inoperable CRE boundary

As such, if a parameter is found to be outside its limit, it is reasonable to allow time to return the value to within its associated limits (allowable time is dependent on the safety significance of the associated parameter), consistent with Standard Technical Specifications.

Regarding the current time of 48 hours, the NRC's Safety Evaluation for License Amendment 170 (Adams Accession No. ML062790364), documented the following (emphasis added):

> The licensee stated that the whole body dose consequences for the Steam Generator Tube Rupture (SGTR) and Main Steam Line Break (MSLB) accidents, as documented in the Updated Safety Analysis Report Tables 15.6-5, 15.6-5A, and 15.1-4, are a smaller percentage of the applicable acceptance criteria (i.e., 10 CFR Part 100) than the thyroid body doses and, therefore, the Completion Time (CT) for restoring the noble gas-specific activity to within the dose equivalent Xenon (DEX) limit in the new Required Action B.l should not be less than the CT for restoring the iodine-specific activity to within the Dose Equivalent Iodine (DEI) limit in Required Action A.2. The required Action A.2 and CT are not being changed in this amendment. Therefore, **the NRC staff concludes that the proposed CT of 48 hours for the new Required Action B.l is acceptable because it provides a reasonable time consistent with the TSs to restore the noble gas-specific activity within the DEX limit.**

Additionally, in a response to a similar question by the Ginna Power Plant (NRC ADAMS Accession No. ML16105A243), it was documented that the DEX would need to increase substantially in order to exceed regulatory limits as its contribution is less than approximately 10% of the total dose for an MSLB or SGTR accident. Therefore, it is reasonable to conclude that the results of an MSLB or SGTR are relatively insensitive to initial DEX activity compared to the DEI activity.

Therefore, consistent with LCO 3.5.4, LCO 3.6.5, and LCO 3.7.10, due to the applicable accidents being relatively insensitive to the DEX initial activity, if the DEX activity exceeds the TS limit, it is reasonable to allow time for a plant to return the DEX activity to within limits rather than requiring an immediate reactor shutdown. As 48 hours has previously been approved for use and thus a reasonable assurance that adequate protection will be maintained has been made with no other changes, this time remains an appropriate completion time.

A supplemental response to this request is provided by Westinghouse in Enclosure I.

RAI ARCB1-SGTR-6- *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).*

Response: Regulatory Guide 1.194 defines infiltration as the following (emphasis added):

> *Infiltration (or Inleakage): The transport of released radioactive materials into the CRE via interstices in the structures, systems, and components that comprise the CRE.* ***Such a transport is driven by pressure differentials between the CRE and areas external to the CRE.***

Regulatory Guide 1.194 refers to Table H-1 NEI 99-03, "Control Room Habitability Guidance" for guidance on infiltration pathways. The following potential vulnerabilities are listed in Table H-1:

- Control Room Ventilation System Operation
- Control Room Ventilation System Integrity
- Other Ventilation Systems
- Penetrations in the Envelope Boundary
- Envelope Doors
- Isolation Dampers
- Other Non-HVAC Systems in the Envelope
- General Boundary Construction

In addition to Table H-1, Table I-2 in NEI 99-03 contains guidance on how to test specific vulnerable components:

- CRE ceiling/roof
- CRE walls

- CRE floor
- CRE penetration in roof/ceilings; walls; floor
- CRE doors
- Electrical conduits
- Ducting, housings located outside the CRE
- Isolation dampers located outside the CRE and the ducting between the CRE wall/floor/ceiling and the damper
- Ducting, housings located within the CRE
- Isolation dampers within the CRE and the ducting between the CRE wall/floor/ceiling and the damper
- Ducting passing through the CRE that is not isolated and is not part of the CR HVAC
- Other systems

The SNUPPS design uses a unique Control Room/Control Building design as the Control Room is inside a mostly-filtered Control Building Envelope (CBE). Letter WO 05-0003 provided the following description of the CRE in regards to the CBE (emphasis added):

> *The WCGS/SNUPPS (Wolf Creek Generating Station/Standardized Nuclear Unit Power Plant System) Control Room Envelope (CRE) design is unique.* ***The control building by and large surrounds the CRE****. The CRE is required by Technical Specifications to be at a positive pressure with respect to its surrounding environment. The Control Building is also designed to be at a positive pressure with respect to its surrounding environment although not positive with respect to the CRE. In the emergency pressurization and filtration mode, the Control Room air volume receives air through a filtration system that takes a suction on the Control Building. The Control Building in turn receives filtered air from the outside environment.*

Table I-2 of NEI 99-03 lists the test that should be performed for each vulnerable component. Several of the components contain the following information regarding how they should be tested: "Not required as positive pressure precludes in-leakage." This note applies to the following:

- CRE ceiling/roof
- CRE walls
- CRE floor
- CRE penetration in roof/ceilings; walls; floor
- CRE doors

Since WCGS uses a positive pressure CRE, in-leakage will be precluded through these sources.

Regarding electrical conduits, these vulnerabilities are also exempted from inleakage testing per NEI 99-03 guidance as long as the conduits are properly sealed. For Wolf Creek, this requirement is met.

For the Other Systems vulnerability, the guidance states that no testing is required if it can be shown that lines in these systems do not leak. Periodic assessments are performed to assure that the CRE is maintained within the licensing and design bases. If an air leak is detected, T.S. 3.7.10 Required Action B.1 and B.2 would require isolation of any air leaks until they are fixed. Thus, this requirement is met.

The CRE only contains control room HVAC ducting, and thus the "Ducting passing through the CRE that is not isolated and is not part of the CR HVAC" vulnerability is not applicable.

Therefore, per NEI 99-03 guidance, the remaining vulnerabilities need to be considered:

- Ducting, housings located outside the CRE
- Isolation dampers located outside the CRE and the ducting between the CRE wall/floor/ceiling and the damper
- Ducting, housings located within the CRE
- Isolation dampers within the CRE and the ducting between the CRE wall/floor/ceiling and the damper

The four vulnerabilities are associated with the CRE ducting, housing, and isolation dampers. The CRE HVAC ducting is located primarily within the CRE and CBE. However, during normal HVAC operation, a supply fan pulls from the normal HVAC intake, which is located on the roof of the auxiliary building. Thus, inleakage is considered from this source. Following a CRVIS, the supply fan is de-energized and the intake is switched from outside air to the control building. However, the dual intake dampers are isolated on a Safety Injection Signal (SIS), and thus are not closed when a CRVIS is present without a SIS. While it is unlikely for inleakage to continue originating from the normal supply location (the supply fan discharges to the CRE, which would be pressurized and isolated during a CRVIS lineup and thus the direction of the flow would be from the CRE), the analyses conservatively do not credit a switch from the normal intake to the emergency intake until the supply dampers are isolated on a SIS.

For emergency operation, the intake ducting for the control room is within the control building. Since the control building emergency HVAC intake is outside air, the dispersion factor from the control building intake is carried through from the control building to the control room for use in the analyses.

Consistent with NEI 99-03 guidance, in-leakage is precluded for vulnerabilities other than ducting, housing, and isolation dampers for WCGS's control room/control building design. Therefore, only these vulnerabilities are considered. The specific intake considered (normal versus emergency) is based upon the applicable HVAC lineup.

The atmospheric dispersion factors modeled for the CRE unfiltered inleakage are limiting as they conservatively account for the bounding path from the environment to the control room. Specifically, for the normal HVAC lineup, the only direct path from the CRE to the outside environment is through the normal HVAC duct intake and thus that is the location modeled. For the emergency HVAC lineup, there is no direct communication from within the CRE to the outside environment. In order to conservatively model inleakage, the control building HVAC duct intake is carried forward for the CRE inleakage, with no credit for mixing within the control building envelope.

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

RAI ARCB1-MSLB-2 *- MSLB*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

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

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

Response: The analysis documented in Section 4.3.10 of Enclosure IV is for a waste gas decay tank failure. Per assumption C.1.a and C.1.b of Regulatory Guide (RG) 1.24, the waste gas decay tank failure analysis is for the tanks utilized to decay the primary coolant radioactive gases. While Figure 11.3-2 does show potential gaseous release points, these release sources are not solely for the decay of primary coolant radioactive gases. Rather, Figure 11.3-2 lists all potential gaseous releases. For example, while the Laundry and Hot Shower Tank is listed in Figure 11.3-2, it is not utilized as a primary coolant decay tank. Additional information for the major sources of gaseous releases is documented in Table 11.1A-3 of the USAR. As shown in Table 11.1A-3, the only holdup tank for a major source of gaseous release is for the gaseous radwaste system (the containment entry is for the atmosphere, not for a tank).

Regarding Wolf Creek Generating Station's gaseous radwaste storage system, as stated in Section 11.3.2.1 of the USAR, "All of the equipment is located in the radwaste building." Since all of the gaseous radwaste storage tanks used to decay primary coolant radioactive gases are located within the radwaste building, it is appropriate to model a radwaste building release for the failure of a waste gas decay tank.

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

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

The NRC question and Westinghouse response to this request is provided in Enclosures I and II.

RAI ARCB1-WT-5*- Waste Gas Decay Tank*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

<u>RAI ARCB1-CREA-2</u>- Control Rod Ejection Accident (CREA)

The NRC question and Westinghouse response to this request is provided in Enclosure I.

<u>RAI ARCB1-CONTROL ROOM-1</u> - All DBAs

The NRC question and Westinghouse response to this request is provided in Enclosure I.

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

<u>RAI ARCB1-CONTROL ROOM-3</u>- Control Room

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.

Response: It should be noted that the radiation dose analysis received for the ingress and egress to the control room is not part of the current licensing basis for WCNGS. Further, no regulatory guidance exists for the calculation of this dose. As such, as discussed during the RAI acceptance phone call on November 21, 2017 with the NRC staff, the analysis discussed below will make reasonable assumptions, consistent with other licensees, to determine the dose. When regulatory guidance is provided, WCGS will review it and compare to the analysis below. A subsequent determination will then be made in regards to making this analysis part of the WCGNS licensing basis.

While responding to ARCB1-CONTROL ROOM-3, it was identified that the value utilized for the Refueling Water RWST backleakage was overconservative. Therefore, a more accurate value for the parameter was developed. Specifically, the past three performances of the surveillance procedure that measures the leakage were reviewed. The following cumulative leakage rates are documented within the completed procedures:

0.16 gpm for procedure completed on 04/25/14
0.24 gpm for procedure completed on 04/06/15
0.14 gpm for procedure completed on 11/08/16

Based upon the previous test results, it is reasonable to model a leakage rate of 2.0 gpm for the RWST backleakage parameter. The surveillance criterion will be updated to reflect the new limit for future performances in order to support implementation of AST.

A supplemental response to this request is provided by Westinghouse in Enclosure I.

<u>RAI ARCB1-CONTROL ROOM-4</u>- Control Room and TSC

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.*

Response:

1) Unfiltered inleakage for ingress and egress into the control room is considered for the revised radiological analyses. However, since the SNUPPS design uses a unique Control Room/Control Building design, an alternate method of integrated inleakage testing is utilized that directly measures the inleakage including contributions due to ingress and egress. Thus, the contribution from ingress and egress (may be more or less than 10 cfm) as well as the unfiltered inleakage is directly measured and accounted for within the dose analyses. This method was presented to the NRC within ET 07-0045 (ADAMS Accession No. ML072770291) as part of WCNOC's response to Generic Letter 2003-01. As discussed in Section 8 of Enclosure II of ET 07-0045 (emphasis added):

> *The objective of the [Atmospheric Tracer Depletion] ATD testing is to demonstrate that unfiltered in-leakage into the Control Room Envelope (CRE) are below the design basis used in the dose assessment calculations. The ATD tests are to be performed in the least intrusive fashion possible while meeting this objective. For SNUPPS-designed plants,* ***this means allowing for normal Control Building (CB) and CRE ingress/egress****. The test is done with a secondary objective to obtain a worst-case CB and CRE configuration and adjacent-zone HVAC operation to provide an upper bound on unfiltered in-leakage. This latter objective, however, is controlled by the plant operators - such that test conditions are consistent with the limiting conditions in their licensing basis. The multi-tracer ATD method meets these objectives.*

As described in ET 07-0045, one value is determined for the total control room envelope inleakage. The resulting value reflects both any unfiltered inleakage from ingress and egress which is allowed during the testing as well as any control room boundary unfiltered inleakage.

The alternate method of integrated inleakage testing was reviewed by the NRC as part of WCNOC's response to Generic Letter 2003-01 (ADAMS Accession No. ML073300464, emphasis added):

> *Although the correlation testing described did not consider a configuration with the unique Control Building/Control Room design of WCGS and how non-conservative an ASTM E 741 method test result might be relative to the ATD method used at WCGS,* ***the NRC staff reviewed the information you provided and determined that for your design of Control Building pressurization/filtration and Control Room pressurization/filtration, the ATD method test as you described is consistent with the guidance of RG 1.197 and the NRC staff concludes that the ATD method is acceptable for WCGS.***

Additionally, the following was stated in the Safety Evaluation for WCNOC's subsequent License Amendment for implementation of TSTF-448, Revision 3 (ADAMS Accession No. ML083390833, emphasis added):

The NRC staff reviewed the licensee's proposed alternative method for measuring CRE inleakage to ensure it meets the criteria for such methods given in Regulatory Guide 1.197. By letter dated November 30, 2007 (ADAMS Accession No. ML073300464), the NRC staff informed WCNOC that the alternative method used by WCGS (Brookhaven National Laboratory-developed Atmospheric Tracer Depletion (ATD) test) is consistent with the guidance of Regulatory Guide 1.197, and the staff concluded that the ATD method is acceptable for WCGS. The staff wrote:

> *In your September 28, 2007, letter, you provided the 12 types of information identified in Regulatory Guide (RG) 1.197, "Demonstrating Control Room Envelope Integrity At Nuclear Power Reactors," Regulatory Position C.1.3 as being needed for the staff to assess the capability and thus acceptability of an alternate test method for determining your CRE inleakage. In this information you included details of testing performed at another nuclear plant that demonstrated a correlation in results between ASTM E741 method and ATD method tests that met the comparability standard of RG 1.197, Regulatory Position 1.2. Although the correlation testing described did not consider a configuration with the unique Control Building/Control Room design of WCGS and how non-conservative an ASTM E741 method test result might be relative to the ATD method used at WCGS, the NRC staff reviewed the information you provided and determined that for your design of Control Building pressurization/filtration and Control Room pressurization/filtration, the ATD method test as you described is consistent with the guidance of RG 1.197 and the NRC staff concludes that the ATD method is acceptable for WCGS.*

The NRC staff was concerned that the ADT test method used at WCGS has not been independently verified or endorsed by one or more industry standards. ***However, on the basis of additional review of the details the NRC staff finds that the proposed alternative method satisfies the criteria of RG 1.197. Therefore, the proposed CRE inleakage measurement SR is acceptable.***

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

The NRC question and Westinghouse response to this request is provided in Enclosure I.

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

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).

Response: The proposed change for this request is provided in Attachments II and III. Attachment II and III provide the Proposed Technical Specification Changes (Mark-up) and Revised Technical Specification Page, respectively.

RAI ARCB1-GENERAL-1- *Dose Conversion Factors*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

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.*

Response: Conservative flow rates were utilized in the radiological consequences analyses. The unfiltered inleakage of 50 cfm is a primary contributor to the overall operator dose. As such, the HVAC flowrates utilized in the analyses were modeled in a way to maximize the contribution of the unfiltered inleakage.

Regarding the safety related HVAC equipment, TS 5.5.11a, b., and f. list specific flow rates with an acceptable range of ±10%.

For the Control Room Emergency Ventilation System – Pressurization flow rate, this is the flow rate pulled from the outside environment into the control building. The air is filtered prior to being discharged to the control building. In addition to the pressurization flow, an unfiltered inleakage of 400 cfm to the control building is also modeled in the analysis. Pressurization flow and unfiltered inleakage are balanced by exhaust flow in the analyses, and exhaust flow acts to purge the radioactivity from the control building. Due to the 95% filtration of the pressurization flow compared to the 0% filtration of the unfiltered inleakage to the control building, the pressurization flow is minimized in order to reduce the purging of the radioactivity and in turn

maximize the dose. Thus, the nominal value of 750 cfm listed in TS 5.5.11a., b., and f, is decreased by 10% for a total flow rate of 675 cfm (as listed in Table 4.3-5 of Enclosure IV).

For the Control Room Emergency Ventilation System – Filtration flow rate, this flow rate is a combination of recirculation from the control room and makeup flow from the control building. Additionally, exhaust flow acts to purge the remaining radioactivity from the control room. As the filtration flow rate will reduce the radioactivity in control room atmosphere via filtration and exhaust, it is minimized in order to maximize the dose. Thus, the 2000 cfm value listed in TS 5.5.11a., b., and f is decreased by 10% and the resulting value of 1800 cfm is used in the analyses. The 1800 cfm is composed of 1250 cfm recirculation and 550 cfm makeup, as listed in Table 4.3-5 of Enclosure IV.

Regarding the Auxiliary/Fuel Building Emergency Exhaust flow rate values listed in TS 5.5.11a., b., and f, these flow rates are not directly modeled in the licensing basis analyses. Rather, it is assumed that any release to the auxiliary building is immediately available for release to the environment. This is a conservative assumption as no credit for dilution is taken within the analysis prior to release.

Regarding the non-safety related HVAC flow rates, these correspond to equipment that is not credited within the safety analyses; rather, the equipment is only modeled in order to make the event more limiting. As filtration within these systems is not credited, a larger value than actual is conservative as it increases the radioactivity brought into the control room making a sensitivity analysis unnecessary.

As discussed in response to ARCB-RAI-16 within Enclosure VI, the non-safety related flow rates modeled within the analyses are greater than 10% of the actual normal flow rates measured within the plant. The 10% variation was applied conservatively and aligns with the flow allowances of both the Control Room Emergency Ventilation Filtration and Control Room Emergency Ventilation Pressurization Systems. Thus, as the non-safety related equipment is not credited within the event, and since the analyses model a more limiting flow rate than the installed plant equipment, the analyses conservatively address the non-safety related HVAC flow rates.

In summary, the safety related flow rates modeled in the analyses conservatively apply the allowable range listed in TS 5.5.11a, b., and f. The non-safety related flow rates modeled in the analyses are greater than actual plant performance and thus are bounding. Additionally, it noted that the flow rates modeled for the AST analyses are consistent with the CLB values currently listed Table 15A-1 of the USAR.

RAI ARCB1-GENERAL-3- *All DBAs*

The NRC question and Westinghouse response to this request is provided in Enclosure I.

PRA Operations and Human Factors Branch (APHB)

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

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

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

Response: Regarding the action of maintaining Steam Generator (SG) water level, the intent of the statement in Section 4.3.6.2.2.2 of Enclosure IV, was to document that sufficient water would be available within the SG to cover the top of the U-tubes. During the emergency response procedures, operators will continuously monitor the SG level to ensure sufficient inventory is available (between 6% and 50% level as indicated on the SG Narrow Range (NR)); however, as auxiliary feedwater flow will automatically initiate on low SG water level (TS Table 3.3.2-1 Function 6.d), two hours is a conservative time to credit the SG inventory. In an actual event the SG water level would automatically reach this level (greater than 6% NR) without operator action within a couple of minutes or reaching the SG low setpoint.

Once the minimum SG water level is reached, due to the availability of makeup secondary water, rather than actively maintaining the level, the operator will simply monitor that the SG level remains between 6% and 50%. As there is not a single failure for the control rod ejection event that will result in a loss of the safety function of auxiliary feedwater, the SG water level will remain above 6% unless multiple failures are considered (i.e., a beyond design basis event). As Enclosure IV is for the licensing basis AST analyses, beyond design basis events are not considered.

Therefore, the basis for crediting the revised mass after two hours relies upon the fact that during the design basis event, auxiliary feedwater will automatically (i.e., without operator action) provide sufficient secondary makeup water to ensure that the SG water level is greater than 6% NR. Since the emergency response procedures do contain a step to continuously monitor that SG water level is between 6% and 50% on the NR scale, and since the response needed by an operator is minimal (unless multiple failures are considered), WCGS considers this action part of the proceduralized response within the Emergency Response Guidelines and not a Time Critical Action.

This justification was provided to the NRC reviewer during a clarification call for this RAI. While this RAI is not being answered as written (i.e., a human factors assessment is not being provided), the approach to provide additional background information was agreed upon as it clarifies why the SG inventory credited is appropriate and conservative.

***APHB-2** Control Room Modeling – Several Accidents*

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

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

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

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

Response: While the operators would have the capability to manually complete the control room isolation during an event, no licensing basis analyses credit manual operator action to complete the control room isolation. Besides two accidents (subsequently discussed), the analyses do not consider the control room being isolated by manual action.

The two accidents that consider manual actuation of the control room are the main steam line break analysis (analysis was updated as part of the response to ARCB1-MSLB-2) and the fuel handling accident within containment with pathways to the auxiliary building open (analysis was originally documented within ET 17-0011 in response to an acceptance review question and two follow-up questions, ARCB1-FHA-5 and ARCB1-FHA-6, are responded to within this letter).

As discussed in the response to ARCB1-MSLB-2, the most limiting time to transfer from the control room HVAC normal lineup to the control room HVAC emergency lineup is at the end of the blowdown phase of the event. This is due to the fact that the radionuclides are already inside the control room and the emergency lineup HVAC flow rates, which are lower than the normal lineup HVAC flow rates, will result in a slower purging of the control room. As such it is conservative to model a manual action to actuate Safety Injection (which will result in a CRVIS and control room isolation) concurrent with the end of the blowdown phase of the event (for this case, 2 minutes).

It is reasonable to model a manual SI, after the P-11 manually blocks an automatic SI signal. From NSAL-02-14,

> *SI is still available and could be automatically (via containment pressure signals) and/or **manually actuated by operators** should conditions warrant such actions. SLB events severe enough to challenge applicable safety limits should provide sufficient indications (i.e., cooldown rates would be much higher than planned) to the operators that such actions are necessary. Westinghouse documentation (PLS) to plants states that **when the SI signals are manually blocked, "The operator must be prepared to manually actuate safety injection if required."***

If a time longer than that of the blowdown phase of the event was modeled, the resulting doses are less limiting due to the higher normal HVAC flow rates purging the control room. Regarding times longer than several hours, as discussed in the response to ARCB1-MSLB-2:

Therefore, as the analysis conservatively models a manual actuation of SI (consistent with NSAL-02-14) to occur at the limiting time of event (isolation occurs concurrent with the end of blowdown), WCGS does not consider this a credited Time Critical Action. Rather, the modeling is done in this manner in order to bound the limiting time when the action could possibly occur, (i.e., a longer time, within reason, would result in a less limiting analysis).

For the FHA described in ET 17-0011 (scenario was analyzed in response to an acceptance review question), this scenario (containment release into auxiliary building through personnel hatch) is not typically analyzed as a licensing basis event. Additionally, as discussed in the response in ET 17-0011, the new scenario analyzed is less limiting than the licensing basis event that models a direct release to the environment. As such, WCGS considers this a nonlimiting scenario that was analyzed to support an acceptance review question; therefore, WCGS is not intending on making the scenario its licensing basis event.

As discussed in the response contained within ET 17-011,

> Also, switchover from normal mode HVAC operation to emergency mode HVAC operation in both the auxiliary building and control room is assumed at 30 minutes after event initiation. This 30-minute time is significantly longer than the generally accepted time of 10 minutes, consistent with Section 3.1.2.d of the Wolf Creek USAR, to identify an event and manually switch to emergency mode. In emergency mode, the control room is pressurized such that inleakage from the auxiliary building to the control room is terminated.

For the FHA within containment, the control room will be in constant communication with the refueling team. Thus, if an FHA were to occur, the control room operators would quickly be alerted and would utilize the appropriate procedure to respond to the event. Since CRVIS can be actuated with a handswitch, and since the control room is in constant communication with the refuel team, it is reasonable to assume that this action will be completed within 10 minutes of a dropped fuel assembly. Furthermore, unlike an accident that results in a reactor trip, for a FHA there will be no time required to diagnose the event due to direct communication with the refueling team and thus the operators can immediately take the necessary actions. Additionally, in order to demonstrate the non-limiting nature of this event, significant margin was applied to inputs. Specifically for this input, 200% margin was applied to the 10 minute time in order to obtain a time of 30 minutes.

Thus, due to the fact that: 1) the scenario discussed within ET 17-0011 is nonlimiting compared to the licensing basis event, 2) there are limited actions required to be taken by a control room operator in order to respond to the event, and 3) significant margin was applied to the already reasonable time of 10 minutes, WCGS does not consider this action a Time Critical Action.

<u>Technical Specifications Branch (STSB)</u> STSB-1

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

Response: Wolf Creek approved a 10 CFR 50.59 evaluation on calculation AN-04-015 revision 1, Radiological Consequences of a Fuel Handling Accident, on April 27, 2006. This calculation evaluated an earlier start of fuel movement following reactor shutdown, specifically, the radiological consequences of a fuel handling accident at 76 hours instead of 100 hours after shutdown. This change affected the assumed initial activities in the radiological consequences analysis of a fuel handling accident as described in USAR Section 15.7.4. Calculation AN-04-015 revision 1 showed that the radiological consequences of a postulated fuel handling accident in the reactor building or the fuel building remain well within the guideline values of 10 CFR 100.

"Well within" means 25% or less of the 10 CFR 100 exposure guideline values, i.e., 75 rem for the thyroid dose and 6.25 rem for the whole-body dose. The resultant doses increased slightly from the previous analyses. However, the increases are less than 10 percent of the difference between the current calculated dose values and the regulatory guideline values. Therefore, the change did not result in more than a minimal increase in the consequences.

References:

1. WCNOC Letter ET 17-0001, "License Amendment Request for the Transition to Westinghouse Core Design and Safety Analyses," January 17, 2017. ADAMS Package Accession No. MLML17054C103.

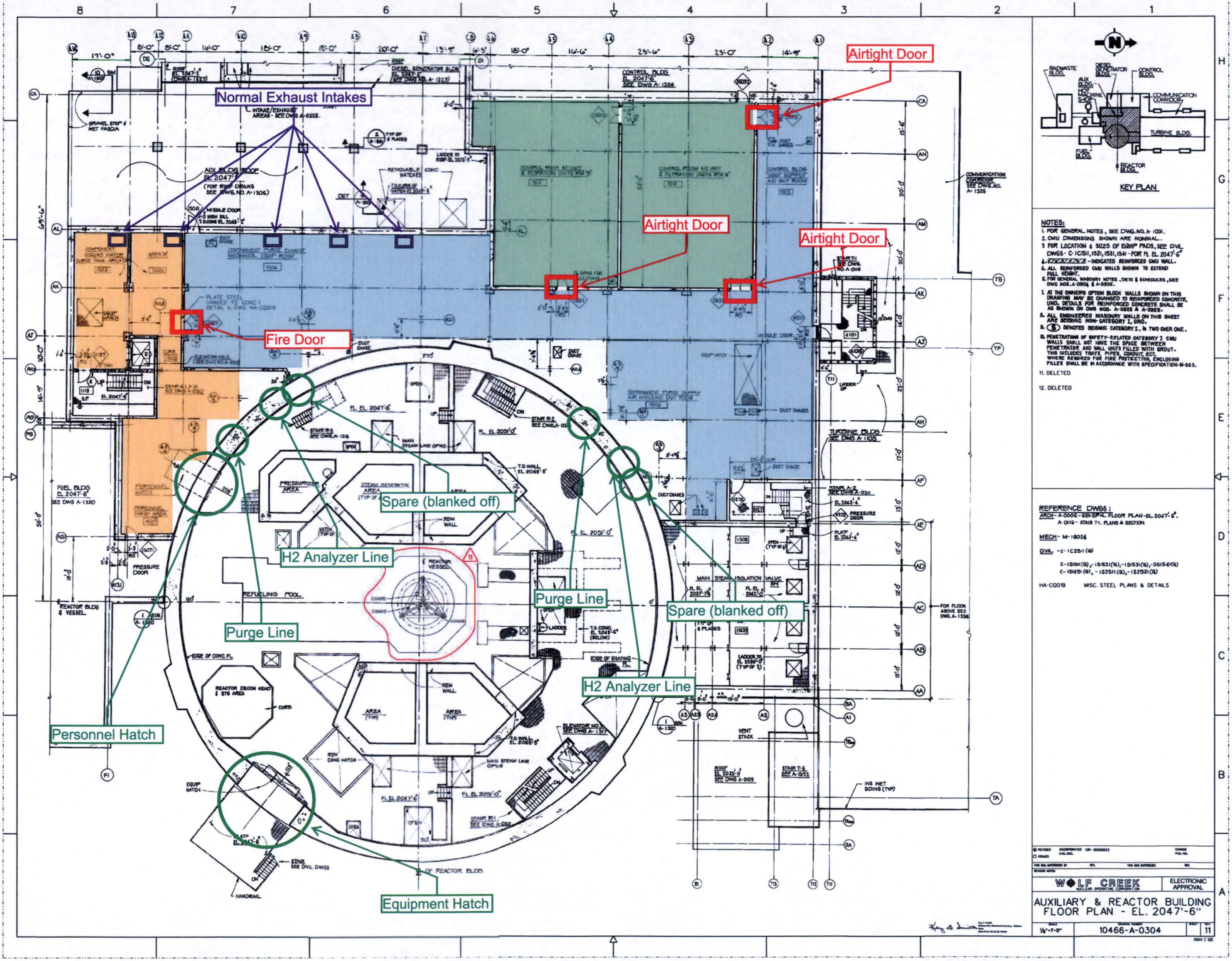
2. Letter from B. K. Singal, USNRC, to A. C. Heflin, WCNOC, "Request for Additional Information Re: License Amendment Request for Transition to Westinghouse Core Design and Safety Analyses Including Adoption of Alternative Source Term," December 4, 2017. ADAMS Accession No. ML17331A178.

**Enclosures to Response to Request for Additional Information
RAI ARCB1-FHA-5 – FHA**

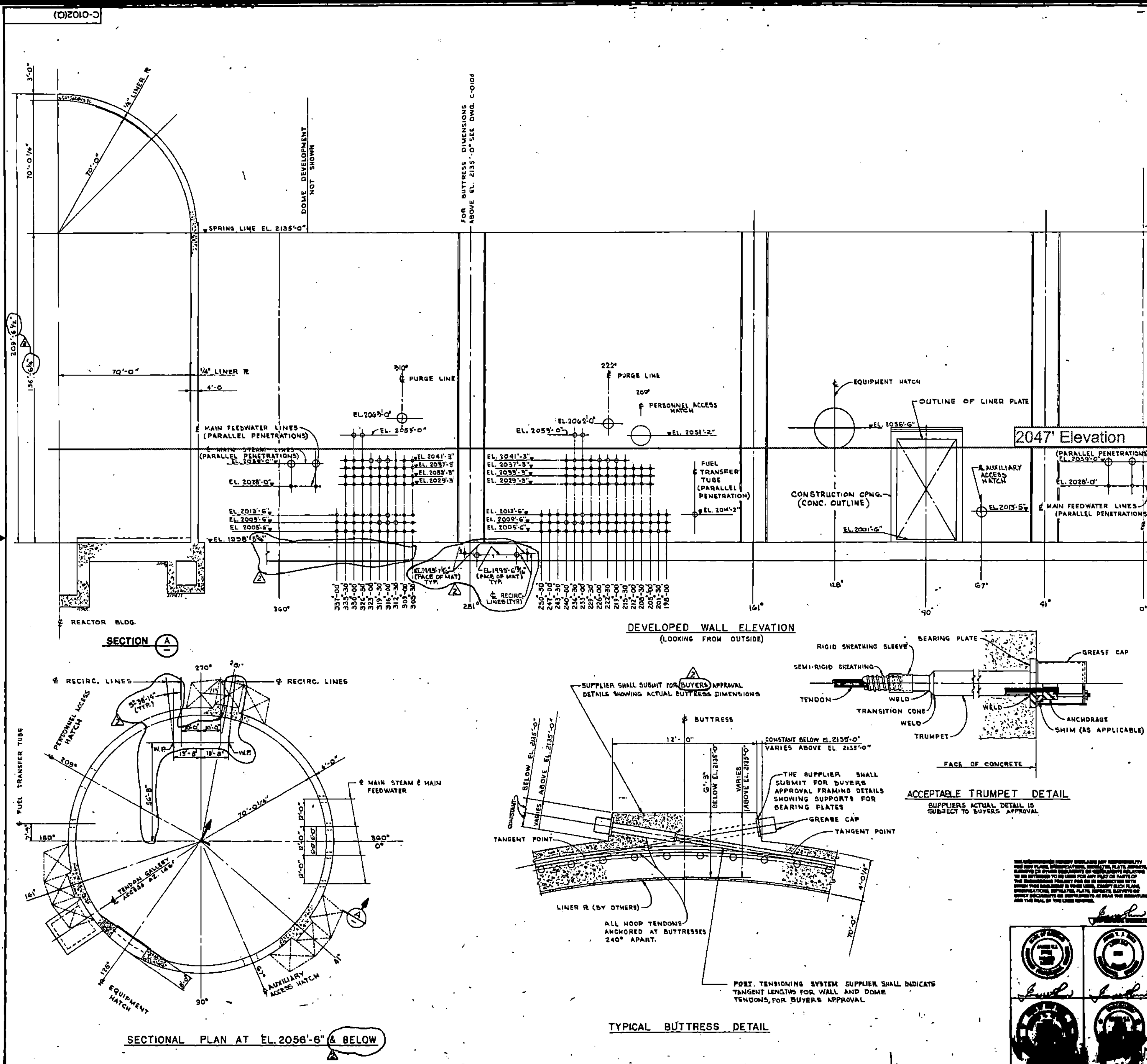
Enclosure I provides a marked up floor plan of the relevant auxiliary building level

Enclosure II provides a diagram showing all containment penetration locations

Enclosure I to Attachment I
(2 pages)


Airtight Door
Normal Exhaust Intakes
Airtight Door
Airtight Door
Fire Door
Spare (blanked off)
H2 Analyzer Line
Purge Line
Purge Line
Spare (blanked off)
H2 Analyzer Line
Personnel Hatch
Equipment Hatch
KEY PLAN
NOTES:
1. FOR GENERAL NOTES, SEE DWG.NO. A-1001.
2. CMU DIMENSIONS SHOWN ARE NOMINAL.
3 FOR LOCATION & SIZES OF EQUIP PADS, SEE CIVIL DWGS - C-1C1511, 1521, 1531, 1541 - FOR FL EL 2047'-6"
4. - INDICATES REINFORCED CMU WALL.
5. ALL REINFORCED CMU WALLS SHOWN TO EXTEND FULL HEIGHT.
6. FOR GENERAL MASONRY NOTES, DETS & SCHEDULES, SEE DWG NOS. A-0904 & A-0905.
7. AT THE OWNER'S OPTION BLOCK WALLS SHOWN ON THIS DRAWING MAY BE CHANGED TO REINFORCED CONCRETE, UNO. DETAILS FOR REINFORCED CONCRETE SHALL BE AS SHOWN ON DWG NOS. A-0928 & A-0929.
8. ALL ENGINEERED MASONRY WALLS ON THIS SHEET ARE SEISMIC NON-CATEGORY I, UNO.
9. (S) DENOTES SEISMIC CATEGORY I, Is TWO OVER ONE.
10. PENETRATIONS OF SAFETY-RELATED CATEGORY I CMU WALLS SHALL NOT HAVE THE SPACE BETWEEN PENETRATOR AND WALL UNITS FILLED WITH GROUT. THIS INCLUDES TRAYS, PIPES, CONDUIT, ECT. WHERE REQUIRED FOR FIRE PROTECTION, ENCLOSING FILLER SHALL BE IN ACCORDANCE WITH SPECIFICATION-M-663.
11. DELETED
12. DELETED
REFERENCE DWGS:
WOLF CREEK
ELECTRONIC APPROVAL
AUXILIARY & REACTOR BUILDING FLOOR PLAN - EL. 2047'-6"
10466-A-0304
11

Enclosure II to Attachment I
(2 pages)



NOTES

1. FOR GENERAL NOTES SEE DWG. C-0101.

REFERENCE DRAWINGS

- C-OL2902 - REACTOR BUILDING LINER PLATE DEVELOPED ELEVATION SH.1
- C-OL2903 - REACTOR BUILDING LINER PLATE DEVELOPED ELEVATION SH.2
- C-OL2904 - REACTOR BUILDING LINER PLATE PENETRATION SCHEDULE SH.1
- C-OL2905 - REACTOR BUILDING LINER PLATE PENETRATION DETAILS SH.1
- C-OL2906 - REACTOR BUILDING LINER PLATE PERSONNEL ACCESS HATCH
- C-OL2907 - REACTOR BUILDING LINER PLATE EQUIPMENT HATCH
- C-OL2908 - REACTOR BUILDING LINER PLATE FLOOR & WALL DETAILS
- C-OL2909 - REACTOR BUILDING LINER PLATE FLOOR BEAM BRKTS.
- C-OL2910 - REACTOR BUILDING LINER PLATE MISC BRACKET DETAILS
- C-OL2912 - REACTOR BUILDING LINER PLATE DOME SECTION
- C-OL2913 - REACTOR BUILDING LINER PLATE DOME DETAILS
- C-OL2915 - REACTOR BUILDING LINER PLATE POLAR CRANE BRACKET DETAILS
- C-OL2916 - REACTOR BUILDING LINER PLATE PENETRATION SCHEDULE SH.2
- C-OL2811 - REACTOR BUILDING LINER PLATE DOME PLAN
- C-OL2812 - REACTOR BUILDING LINER PLATE DOME PLAN-STIFFENERS
- C-OL2918 - REACTOR BUILDING LINER PLATE PENETRATION DETAILS SH. 2

MATERIAL RESPONSIBILITY

1. POST-TENSIONING SYSTEM FURNISHED UNDER SPEC. 10466-C155.
2. TENDON SHEATHING FURNISHED UNDER SPEC. 10466-C154.
3. TENDON SHEATHING FILLER MATERIAL FURNISHED UNDER SPEC. 10466-C157.

NO.	DATE	REVISIONS	BY	CHK
2		REVISED AS NOTED AND ISSUED FOR CONSTRUCTION	R	
1		ISSUED FOR FABRICATION	H	
0		ISSUED FOR BIDS	HCP	

SCALE N.T.S. DATE DESIGNED DRAWN HCP

BECHTEL
GAITHERSBURG, MARYLAND

SNUPPS

REACTOR BUILDING PRESTRESSING REQUIREMENTS WALL DETAILS

JOB NO.	BECHTEL DRAWING NO.	REV
10466	C-0102(Q)	2

Proposed Technical Specification Changes (Mark-ups)

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

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, 5, and 6,
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)

3.7 PLANT SYSTEMS

3.7.13 Emergency Exhaust System (EES)

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

LCO 3.7.13 Two EES trains shall be OPERABLE.

-----NOTE-----
The auxiliary building or fuel building boundary may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies in the fuel building.

-----NOTE-----
The SIS mode of operation is required only in MODES 1, 2, 3, and 4. The FBVIS mode of operation is required only during movement of irradiated fuel assemblies in the fuel building.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable to the FBVIS mode of operation.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One EES train inoperable in MODE 1, 2, 3, or 4..	A.1	Restore EES train to OPERABLE status.	7 days
B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1, 2, 3, or 4.	B.1	~~Restore auxiliary building boundary to OPERABLE status.~~	~~24 hours~~

INSERT 3.7-33

(continued)

INSERT 3.7-33

B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1, 2, 3, or 4.	B.1 Initiate actions to implement mitigating actions.	Immediately
	<u>AND</u>	
	B.2 Verify mitigating actions ensure main control room occupants do not exceed 10 CFR 50 Appendix A GDC 19 limits.	24 hours
	<u>AND</u>	
	B.3 Restore building boundary to OPERABLE status.	24 hours

Revised Technical Specification Pages

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

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

3.7 PLANT SYSTEMS

3.7.13 Emergency Exhaust System (EES)

LCO 3.7.13 Two EES trains shall be OPERABLE.

--NOTE--
The auxiliary building or fuel building boundary may be opened intermittently under administrative controls that ensure the building boundary can be closed consistent with the safety analysis.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies in the fuel building.

--NOTE--
The SIS mode of operation is required only in MODES 1, 2, 3, and 4. The FBVIS mode of operation is required only during movement of irradiated fuel assemblies in the fuel building.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable to the FBVIS mode of operation.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One EES train inoperable.	A.1 Restore EES train to OPERABLE status.	7 days
B. Two EES trains inoperable due to inoperable auxiliary building boundary in MODE 1, 2, 3, or 4.	B.1 Initiate actions to implement mitigating actions. AND	Immediately (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Verify mitigating actions ensure main control room occupants do not exceed 10 CFR 50 Appendix A GDC 19 limits.	24 hours
	AND		
	B.3	Restore building boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. OR Two EES trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the fuel building.	D.1	Place OPERABLE EES train in operation in FBVIS mode.	Immediately
	OR		
	D.2	Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two EES trains inoperable due to inoperable fuel building boundary during movement of irradiated fuel assemblies in the fuel building.	E.1 Restore fuel building boundary to OPERABLE status.	24 hours
F. Required Action and associated Completion Time of Condition E not met. OR Two EES trains inoperable during movement of irradiated fuel assemblies in the fuel building for reasons other than Condition E.	F.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each EES train for ≥ 15 continuous minutes with the heaters operating.	31 days
SR 3.7.13.2	Perform required EES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.13.3	Verify each EES train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.13.4	Verify one EES train can maintain a negative pressure ≥ 0.25 inches water gauge with respect to atmospheric pressure in the auxiliary building during the SIS mode of operation.	18 months on a STAGGERED TEST BASIS
SR 3.7.13.5	Verify one EES train can maintain a negative pressure ≥ 0.25 inches water gauge with respect to atmospheric pressure in the fuel building during the FBVIS mode of operation.	18 months on a STAGGERED TEST BASIS

Attachment 1 to SAP-18-2, "Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies" – Non-Proprietary
(61 pages)

Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies [Non-Proprietary]

(60 pages including cover page)

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

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

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

The licensee further stated, in part:

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

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

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

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

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

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

WCGS's LOAC analysis assumes:

> The reactor trip associated with the LOAC creates an iodine spike that is assumed to increase the iodine release rate from the fuel to the RCS [reactor coolant system] to a value 500 times greater than the appearance rate

corresponding to the maximum equilibrium RCS concentration of 1.0 μCi/gm DE I-131...The duration of the accident-initiated iodine spike is assumed to be 8 hours.

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

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

   or

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

Response

2. The LOAC analysis source term is not consistent with RG 1.183, Appendix E, Regulatory Position 2.1. However, as RG 1.183 Appendix E is for an MSLB and not for the Loss of Non-Emergency Alternating Current Power (LOAC), it was not the intent to make the LOAC analysis consistent with RG 1.183, Appendix E. Thus, a pre-accident iodine spike was not modeled for the Loss of Non-Emergency Alternating Current Power (LOAC).

The Standard Review Plan (SRP) for LOAC (NUREG-0800, Section 15.2.6) refers to SRP Section 15.0.3, which provides no specific guidance on the LOAC event. However, ANSI N18.2-1973 identifies both the LOAC event and minor RCS leak event (i.e. small line break or letdown line break) as Condition 2 events. SRP Section 15.0.3 provides guidance on design basis accident radiological consequence analyses for advanced light water reactors, and refers to SRP Section 15.6.2 for guidance on the small line break event. SRP Section 15.6.2 describes an accident-initiated (also called coincident) iodine spike, and does not describe a pre-accident iodine spike. Additionally, SRP Section 15.0.3, Table 1 defines a limit of 2.5 rem TEDE for the Small Line Break; this limit is consistent with other event scenarios that consider an accident-initiated iodine spike, such as Main Steamline Break and Steam Generator Tube Rupture. The pre-accident spike scenarios for those events have a higher limit of 25 rem TEDE. Because the available guidance is silent on a pre-accident spike for a small line break and a LOAC is a Condition 2 event similar to the small line break event, a pre-accident iodine spike was not analyzed for the LOAC event. Additionally, the more restrictive limit of 2.5 rem TEDE was utilized.

RAI ARCB1-LOAC-2 - LOAC

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

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

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

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

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

Response

WCNOC is requesting that the acceptance criteria be that under accident criteria in 10 CFR 50.67 and RG 1.183. An offsite dose acceptance criterion of 2.5 rem TEDE is requested for the Loss of Non-Emergency Alternating Current Power (LOAC) event. The LOAC event is not described in Regulatory Guide 1.183. The Standard Review Plan (SRP) for LOAC (NUREG-0800, Section 15.2.6) refers to SRP Section 15.0.3, which provides no specific guidance on the LOAC event. However, ANSI N18.2-1973 identifies both the LOAC event and minor RCS leak event (i.e. small line break or letdown line break) as Condition 2 events. SRP Section 15.0.3, Table 1 defines an offsite dose limit of 2.5 rem TEDE for the Small Line Break. Because the available guidance is silent on the LOAC and a LOAC is a Condition 2 event similar

to the small line break event, an offsite dose acceptance criterion of 2.5 rem TEDE is requested for the LOAC event.

The dose to operators in the control room is calculated for 30 days following the start of releases and the acceptance limit of 5.0 rem TEDE is defined in 10 CFR 50.67.

Supplement 1 to NUREG-0737 states that the TSC doses will not exceed 5 rem whole body, or its equivalent to any part of the body. GDC 19 equates 5 rem whole body, or its equivalent to any part of the body, to 5 rem TEDE. This 5 rem TEDE limit is applied to the TSC staff for the duration of the accident.

Enclosure IV, Section 4.3.4.3 should be revised to read as follows:

4.3.4.3 Acceptance Criteria

The EAB and LPZ dose acceptance criterion for a LOAC is 2.5 rem TEDE. The LOAC event is not described in Regulatory Guide 1.183. The Standard Review Plan (SRP) for LOAC (NUREG-0800, Section 15.2.6) refers to SRP Section 15.0.3, which provides no specific guidance on the LOAC event. However, ANSI N18.2-1973 identifies both the LOAC event and minor RCS leak event (i.e. small line break or letdown line break) as Condition 2 events. SRP Section 15.0.3, Table 1 defines an offsite dose limit of 2.5 rem TEDE for the Small Line Break. Because the available guidance is silent on the LOAC and a LOAC is a Condition 2 event similar to the small line break event, an offsite dose acceptance criterion of 2.5 rem TEDE is requested for the LOAC event. The control room dose acceptance criterion is 5 rem TEDE per 10 CFR 50.67. The acceptance criterion for the TSC dose is 5 rem TEDE as allowed by GDC 19, in accordance with Reference 2 [Section 4.3.4.5].

RAI ARCB1-LLBA-1 (Letdown Line Break)

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

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

1) Please explain how the flashing fraction was determined. Was the flashing fraction determined consistent with RG 1.183 Appendix A Regulatory Position 5.4 (which uses a constant enthalpy, h, process based on the maximum time-dependent temperature of the water circulating outside the containment: $FF = \frac{h_{f1}-h_{f2}}{h_{fg}}$ where h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation condition; and h_{fg} is the heat of vaporization at 212 degrees Fahrenheit).

Response

The flashing fraction was calculated using the equation from Regulatory Guide 1.183, Appendix A, Position 5.4. The 18% flashing fraction was calculated using the letdown pressure of 600 psig and letdown temperature of 380 °F, as listed in Table 4.3-10 of Enclosure IV.

Flashing fraction = $(h_{f1} - h_{f2})/h_{fg} = 0.18$

Where:

h_{f1} (380°F, 600 psig) = 354.1 Btu/lbm

h_{f2} (14.7 psia) = 180.2 Btu/lbm

h_{fg} (14.7 psia) = 970.3 Btu/lbm

The update of the Letdown Line Break dose analysis (See response to LLBA-4) uses a letdown temperature of 290 °F. This results in a calculated flashing fraction of 0.08.

Flashing fraction = $(h_{f1} - h_{f2})/h_{fg} = 0.08$

Where:

h_{f1} (290°F, 600 psig) = 260.5 Btu/lbm

h_{f2} (14.7 psia) = 180.2 Btu/lbm

h_{fg} (14.7 psia) = 970.3 Btu/lbm

The updated analysis conservatively applied a 10% airborne fraction consistent with Regulatory Guide 1.183, Appendix A, Position 5.5.

RAI ARCB1-LLBA-2 (Letdown Line Break)

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

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

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

Response

A pre-accident iodine spike is not required to be analyzed for the Letdown Line Break event. Radiological consequences for the failure of small lines carrying primary coolant outside containment (i.e. Letdown Line Break) are not described in Regulatory Guide 1.183, but guidance is provided by NUREG-0800, the Standard Review Plan (SRP), Sections 15.0.3 and 15.6.2. SRP Section 15.0.3 provides guidance on design basis accident radiological consequence analyses for advanced light water reactors, and refers to SRP Section 15.6.2 for guidance on the Small Line Break event. SRP Section 15.6.2 describes an accident-initiated (also called coincident) iodine spike, and does not describe a pre-accident iodine spike. Additionally, SRP Section 15.0.3, Table 1 defines a limit of 2.5 rem TEDE for the Small Line Break; this limit is consistent with other event scenarios that consider an accident-initiated iodine spike, such as Main Steamline Break and Steam Generator Tube Rupture. The pre-accident spike scenarios for those events have a higher limit of 25 rem TEDE. Because the regulatory guidance is silent on a pre-accident spike for a small line break, and the SRP provides guidance for the Small Line Break that does not include a pre-accident iodine spike, a pre-accident iodine spike was not analyzed for the LLB event.

RAI ARCB1-LLBA-3 (Letdown Line Break)

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

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

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

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

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

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

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

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

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

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

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

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

Response

The letdown line break analysis assumes the break occurs outside of containment and releases activity into the auxiliary building. The activity is assumed to be instantaneously transported to the Unit Vent Stack with no credit taken for dilution, holdup or cleanup in the auxiliary building. The model is that of a direct atmospheric release with the Unit Vent Stack X/Q applied.

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

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

> Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3). The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.

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

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

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

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

1. Please explain how the removal coefficient(s) were calculated for the WCNOC design and how the assumptions are consistent with RG 1.183. Please provide enough detail (including the aerosol size distribution in containment after the sprays 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.

Response

The LOCA analysis assumed a sedimentation removal coefficient of 0.1 hr^{-1} for aerosols in unsprayed regions of containment and in the sprayed regions of containment after the sprays are terminated at 5 hours. It is also conservatively assumed that sedimentation removal does not continue beyond a DF of 1000.

The Containment Systems Experiments (CSE) are described in Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983. The CSE examined air cleanup through natural transport processes. Figure 4-2 of the IDCOR report provides cesium concentration vs. time in a containment environment, from Run A-5 of the CSE. In Table 4-7 of the IDCOR report, it is reported that a large fraction of the aerosols were deposited on the floor rather than on the walls indicating that sedimentation was the dominant removal process for the test.

In Figure 4-2 of the IDCOR report, it is shown that the airborne cesium concentration, starting at approximately 700 $\mu g/m^3$ is reduced by a factor of 12,400 over a 24 hr time interval. The effective removal rate is therefore:

$$\lambda = -\ln(1/12{,}400)/24\text{ hr}$$
$$= 0.393\text{ hr}^{-1}$$

Even considering the later stages of the test (after 8 hours), when the airborne concentration in the main room has already been reduced by a factor of 70 to a concentration of about 10 $\mu g/m^3$, the concentration is reduced by a factor of about 180 at 24 hours and the effective removal rate over this time period is:

$$\lambda = -\ln(1/180)/16\text{ hr}$$
$$= 0.325\text{ hr}^{-1}$$

The Wolf Creek Cesium Core radionuclide inventories (See Enclosure IV, Table 4.3-1a) are used to demonstrate the conservatism of the 0.1 hr^{-1} sedimentation removal coefficient in unsprayed regions of the containment and in sprayed regions of the containment when sprays are not operating. The mass of a given nuclide is calculated from the activity, the half-life, and the atomic number.

A sample calculation for Cs-134 is as follows:

Activity Remaining (Ci) = Core Inventory (Ci) * Release Frac. / Max Credited Particulate DF

= 1.65E7 Ci * 0.3 / 1000 = 4.95E3 Ci

Activity Remaining (Bq) = Activity Remaining (Ci) * 3.7E+10 Bq/Ci

= 4.95E3 Ci * 3.7E+10 Bq/Ci = 1.832E+14 Bq

Half Life of Cs-134 = 2.062 yr * 365.25 day/yr * 24 hr/day * 3600 sec/hr = 6.50E7 sec

Number of atoms = Activity (Bq) * Half-life (sec) / ln(2)

= 1.832E14 Bq * 6.50E+07 sec / ln(2) = 1.72E+22 atoms

Mass (g) = Number of atoms * Atomic Mass Number (g/mole) / 6.022E+23 (atoms/mole)

= 1.72E+22 atoms * 134 g/mole / 6.022E+23 atoms/mole = 3.83E+00 g

Nuclide	Inventory Remaining in Containment (Ci)		Half-life	Mass
Cs-134	1.65E7 Ci * 0.3 / 1000 =	4.95E3 Ci	2.062 yr	3.83E+00 g
Cs-136	3.951E6 Ci * 0.3 / 1000 =	1.19E3 Ci	13.1 day	1.62E-02 g
Cs-137	1.11E7 Ci *0.3 / 1000 =	3.33E3 Ci	30 yr	3.83E+01 g
Total =				4.21E+01 g

Note that Cs-138 was not included in the above evaluation because its very short half-life, combined with spray removal, will result in an insignificant mass remaining airborne at the end of spray operation.

When uniformly distributed over the maximum free containment volume of 2.7E6 ft^3 (7.64E4 m^3) (See Enclosure IV, Table 4.3-12), this represents a concentration of approximately 550 $\mu g/m^3$ of cesium in the containment atmosphere. As noted above, the CSE tests showed that at an air concentration of 10 $\mu g/m^3$, the sedimentation removal coefficient was above 0.3 hr^{-1}. Thus, the use of a sedimentation removal coefficient of 0.1 hr^{-1} is conservative.

The Shearon Harris Alternate Source Term Safety Evaluation Report (ADAMS Accession No. ML012830516) discusses the applicability of the IDCOR report in comparison to NUREG/CR-6189, which is cited in Regulatory Guide 1.183, and accepted the sedimentation removal coefficient of 0.1 hr^{-1}. The sedimentation removal coefficient of 0.1 hr^{-1} has been also accepted by the NRC for Indian Point Unit 2 (ADAMS Accession No. ML003727500) in addition to the aforementioned Point Beach (ADAMS Accession No. ML110240054), and most recently for Farley (ADAMS Accession No. ML17271A265).

<u>RAI ARCB1-LOCA-2</u> - LOCA Control Room Modeling

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

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

Response

Emergency mode, including filtration of the control room and control building inflow and recirculation flows, is modeled beginning at 120 seconds past the start of the LOCA.

RAI ARCB1-LOCA-4 - LOCA

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

> It should be noted that back-leakage of sump fluid through the RWST [refueling water storage tank] is not considered in the post-LOCA analysis based on the plant emergency procedure for transfer to cold leg recirculation that requires closure of the 24" RWST outlet valve within 16 hours of SI [safety injection] initiation to limit releases from the RWST to the atmosphere. This results in three valve isolation and a minimum of two valve isolation in the long term with a single failure. Refer to USAR Table 6.3-5. Therefore, there is no need to address the potential issue of reduced pH in the RWST, which could lead to a potential radioactive iodine release from the RWST to the environment.

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

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

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

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

Response

1. The analysis accounts for 10% of the iodine released to the RWST in order to bound a potential release to the environment. In order to address the potential issue of a reduced pH within the RWST, the following justification is provided:

The RWST maximum boron concentration of 2500 ppm (from Wolf Creek Technical Specification Surveillance Requirement 3.4.5.3) would have an expected pH of approximately 4.7, based on Table 2-1 of WCAP-17021-NP, Rev. 1 (available via ADAMS Accession No. ML11220A169).

From the Wolf Creek ORIGEN-S calculation results, there are approximately 2.0E4 grams of iodine in the core. The mass of iodine is dominated by stable iodines with I-129 comprising over 70% of the total iodine mass and I-127 comprising over 20% of the total iodine mass. A representative mass number of

129 is used to convert the 2.0E4 grams of iodine to approximately 155 gram-atoms of iodine. Following a LOCA, 40% of this iodine, or 62 gram-atoms is assumed to enter the sump. The sump volume is 460,000 gallons or 1.74E6 liters, resulting in a concentration of approximately 3.6E-5 gram-atoms/L.

NUREG/CR-5950 (ADAMS Accession No. ML063460464), Section 3.2 presents an iodine radiolysis model that predicts the fraction of iodine converting to the elemental form. Using that model with assumptions of a pH of 4.7 and the sump iodine concentration of 3.65E-5 gram-atoms/L, a conversion of approximately 20% is estimated. Note that this neglects any mixing between the sump fluid and the RWST fluid, which would greatly reduce the conversion to elemental iodine, and also neglects any pH increase in the RWST from the alkaline sump fluid.

Additionally, not all of the iodine that converts to the elemental from would be expected to become airborne. The iodine would partition across the surface of the RWST fluid. NUREG/CR-5950 Section 3.3.1 presents a model for calculating the iodine partition coefficient, based on the temperature of the fluid. With an initial RWST temperature of 100°F, the calculated partition coefficient would be greater than 30, i.e. the iodine concentration in the RWST liquid would be greater than 30 times that of the vapor.

The leakage into the RWST is expected to be terminated by 16 hours past the start of the event. The iodine transferred to the RWST during that time would not be a significant dose contributor after accounting for the conversion and partitioning effects.

However, from the NUREG/CR-5950 models it is recognized that as iodine concentration increases, the conversion to elemental iodine increases (assuming no pH increase from the alkaline sump fluid), and that as temperature increases, the partition coefficient decreases. Both of these effects would tend to increase the iodine available for release.

To conservatively bound these effects, the LOCA dose analysis models 10% of the iodine in the leakage into the RWST becoming airborne into the RWST gas space. A minimum expected RWST gas space is conservative as it results in a higher RWST gas space concentration. The airborne iodine is assumed to mix homogenously in the RWST gas space. As the leakage introduces sump fluid into the RWST at a rate of 2.0* gpm, the gas at the top of the RWST is displaced at the same rate, reducing the gas space volume and concentrating the activity in the gas space. Although the leakage is expected to be terminated by 16 hours past the start of the event, the dose analysis conservatively models the continuation of the leakage until 30 days, maintaining the 10% airborne iodine assumption and continuing to reduce the RWST gas space as more liquid is transferred from the sump.

2. The dose analysis models the continuation of the back-leakage until 30 days.

* See the response to ARCB1-CONTROL ROOM-3 where the leakage has been uniformly changed from 3.8 gpm to 2.0 gpm.

RAI ARCB1-FHA-2 - FHA

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

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

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

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

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

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

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

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

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

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

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

Response

WCAP-16072-P-A, Appendix B contains responses to NRC RAIs under the heading "Additional Information Concerning Dose Calculations." Dose Question 2 was related to fuel handling accident fission product release iodine scavenging in the spent fuel pool / reactor cavity. Dose Question 2 noted that the pool decontamination factors in Safety Guide 25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," (a predecessor document to Regulatory Guides 1.183 and 1.195) are predicated on a maximum fuel rod pressurization of 1200 psig. Dose Question 2 also noted that the assumptions in Safety Guide 25 were largely developed from the results of experiments performed by Westinghouse as reported in WCAP-7518-L. The Westinghouse response to Dose Question 2 also utilized the results of experiments reported in WCAP-7518-L and established that the increase in fuel pin pressure from 1200 psig to 1500 psig did not affect the pool decontamination factors in Safety Guide 25. The NRC agreed with this conclusion in their SER on WCAP-16072 and extended the conclusion to Regulatory Guides 1.183 and 1.195 with the statement

> *...the staff has determined that there is reasonable assurance that fuel rod design pressures of up to 1500 psig will not invalidate analysis assumptions related to iodine decontamination. The staff has also determined that this conclusion remains valid for the decontamination factor of 200 provided in RG 1.183 and RG 1.195, which supercede SG 25 for alternative source terms and TID14844 source terms, respectively.*

Similar to Wolf Creek, Comanche Peak is a Westinghouse-designed plant with 17x17 fuel. On August 22, 2005, Comanche Peak submitted a License Amendment Request (ADAMS Accession No. ML052380403) to revise Technical Specifications as part of the response to NRC Generic Letter 2003-01, "Control Room Habitability." As part of the resolution to the Generic Letter, the Comanche Peak dose calculations, including those for the fuel handling accident (FHA), were updated to follow NRC Regulatory Guide 1.195. It is noted that the Comanche Peak FHA included a pool depth less than 23 feet (which is not the case with Wolf Creek) and fuel pin pressures of up to 1500 psig. The SER for the Comanche Peak submittal (ADAMS Accession No. ML070310476) again accepted the use of the WCAP-7518-L experimental data and formulation. Section 3.1.2.1 of the SER states, in part:

> *SRP 15.7.4 states that if factors less conservative than those recommended by RG 1.25 are used, guidance provided by G. Burley, Radiological Safety Branch, Division of Reactor Licensing, NRC, titled, "Evaluation of Fission Product Release and Transport for a Fuel Handling Accident," revised October 5, 1971, should be consulted to determine if an adequate basis for the proposed*

deviation exists. This evaluation is based, in part, on an earlier Westinghouse Topical Report WCAP-7518, "Radiological Consequences of a Fuel Handling Accident," dated June 1970. The basis of RG 1.25 and ultimately RG 1.195 utilizes the experimental data and formulation of WCAP-7518 in a manner to ensure a conservative result appropriate for licensing purposes. The methodology used in WCAP-7518 is similar to that of WCAP-7828 used by the licensee. The NRC staff found the assumptions and methodology used to calculate the SFP DF acceptable and thereby finds the new SFP DF value to be acceptable.

With respect to Fuel Handling Accident guidance, both Regulatory Guide 1.183 and 1.195 are successor documents to Safety Guide 25. A comparison of the Appendix B.2's (entitled "Water Depth") of Regulatory Guides 1.183 and 1.195 shows similar guidance, i.e. the same overall decontamination factor of 200 for iodines, and the same recommendation to use the paper by G. Burley, "Evaluation of Fission Product Release and Transport for a Fuel Handling Accident," in the event that select conditions are not met. With the same overall pool DF, and the same recommendation to use the Burley paper in the event that select conditions are not met, it is apparent that the technical basis for the overall spent fuel pool DF of 200 is the same. This is supported by the SER on WCAP-16072-P-A as quoted above.

As noted in the Comanche Peak SER "*The basis of RG 1.25 and ultimately RG 1.195 utilizes the experimental data and formulation of WCAP-7518 in a manner to ensure a conservative result appropriate for licensing purposes.*" Because Regulatory Guide 1.183 and Regulatory Guide 1.195 have the same technical basis (i.e. the experimental data and formulation of WCAP-7518), the experimental data and formulation of WCAP-7518-L can be used in a manner to ensure a conservative result appropriate for licensing purposes while following Regulatory Guide 1.183.

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The overall DF of 200 in Regulatory Guide 1.183 is based on gap iodine chemical fractions of 0.9985 for elemental iodine and 0.0015 for organic iodine. The overall DF is calculated:

$$DF_{overall} = \frac{1}{\left(\frac{F_{elemental}}{\mathrm{DF}_{elemental}}\right) + \left(\frac{F_{organic}}{\mathrm{DF}_{organic}}\right)}$$

Where

$DF_{overall}$ is the overall DF of 200

$F_{elemental}$ is the fraction of iodine as elemental, 0.9985

$DF_{elemental}$ is the elemental iodine DF

$F_{organic}$ is the fraction of iodine as organic, 0.0015

$DF_{organic}$ is the organic iodine DF of 1

$$200 = \frac{1}{\left(\frac{0.9985}{DF_{elemental}}\right) + \left(\frac{0.0015}{1}\right)}$$

A $DF_{elemental}$ of approximately 285.3 solves the expression. The elemental iodine DF calculated using the experimental data (extrapolated to 1500 psig fuel pin pressure) and formulation of WCAP-7518-L yields an elemental iodine DF well above 285, and thus the overall DF of 200 remains conservative at fuel pin pressures of up to 1500 psig.

<u>RAI ARCB1-FHA-4</u> - FHA

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

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

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

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

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

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

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

Response

Alkali metals (e.g. Cesiums and Rubidium) were considered in the development of the FHA dose analysis, but were not modeled in the FHA dose calculation as they are assumed to be retained by the pool, and thus do not contribute to the dose.

Regulatory Guide 1.183, Regulatory Position 3.5 states, in part:

> *With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs...*

Thus, the alkali metals are assumed to be in particulate form.

Regulatory Guide 1.183, Appendix B, Regulatory Position 3 states, in part:

> *Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).*

Thus, the alkali metals are assumed to be retained by the pool, and thus do not contribute to the dose.

Because the alkali metals do not contribute to the FHA dose, they were not modeled in the FHA dose calculation.

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

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

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

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

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

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

Response

The partition coefficient of 100 is applied only to the radioactivity in the bulk water assumed to become vapor by steaming and not to any primary-to-secondary flow that flashed to steam. After reactor trip, iodine and alkali metals in the flashed portion of the break flow are assumed to be released directly to the environment with no mitigation, dilution, or credit for scrubbing. Throughout the event, iodine and alkali metals in the unflashed portion of the break flow are assumed to mix with the secondary coolant, where they are subject to release via steaming. A partition factor of 100 is applied to the iodines for these steaming releases. The moisture carryover of 0.25% is applied to the alkali metals for these steaming releases. Note that prior to reactor trip, the condenser DF of 100 is credited for iodine and alkali metals released from the SGs, including those in the flashed break flow.

RAI ARCB1-SGTR-2 - SGTR

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

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

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

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

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

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

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

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

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

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

Response

The loss of offsite power is taken to be a consequence of the reactor trip generated by the initiating event, not a second, independent event occurring simultaneously. The SGTR dose analysis utilizes the mass releases and timing from the transient thermal-hydraulic analysis described in Section 2.7.3 of Enclosure I of "Wolf Creek, License Amendment Request for the Transition to Westinghouse Core Design and Safety Analyses" (ADAMS Accession No. ML17054C103). This thermal-hydraulic analysis follows the methodology of Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a SGTR Accident," which includes the assumption of a loss of offsite power due to reactor trip.

SGTR doses with the assumed loss of offsite power concurrent with reactor trip have been accepted by the NRC for Point Beach (ADAMS Accession No. ML110240054, Section 2.1.2.2.2), Shearon Harris (ADAMS Accession No. ML012830516, Section 3.1.2), and Diablo Canyon (ADAMS Accession No. ML17012A246, Section 3.3.7).

Based on the above, the SGTR analysis conforms to the aforementioned regulatory positions.

It should be noted that, for the Wolf Creek analysis, reactor trip occurs at 52 seconds. A loss of offsite power at the start of the event would not significantly affect the plant response, only the timing. Essentially, the sequence of events would shift by approximately 52 seconds. The major dose contributors, such as post-trip flashed break flow, total break flow and ruptured SG releases are driven by decay heat removal and so would be unaffected.

RAI ARCB1-SGTR-3 - SGTR

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

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

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

RG 1.183, Appendix F, Regulatory Position 5.1 states:

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

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

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

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

Response

Primary to secondary leakage is not location-specific, unlike break flow. The leakage would transfer from the primary side to the secondary side, where the iodine and alkali metals would mix with the secondary coolant before being released subject to partitioning (for iodine) or moisture carryover (for alkali metals); noble gases would become immediately airborne and exit the SG. This would be the same for leakage into any SG, ruptured or intact.

The leakage into the intact SG continues, and the activity continues to accumulate, until releases from the intact SGs stop at 43200 seconds. The post-trip intact SG steam releases are approximately 2.04E6 lbm (Table 4.3-11 of Enclosure IV). The major steam releases from the ruptured SG are terminated when the failed-open power-operated relief valve is isolated at 2902 seconds. The post-trip ruptured SG steam

releases are approximately 1.80E5 lbm. Note that a small amount of steam is released from the ruptured SG after break flow termination as the ruptured SG is depressurized to residual heat removal entry conditions.

With the higher releases from the intact SGs over a longer period and with continuing accumulation of activity in the intact SGs resulting in higher release concentrations as the transient progresses, it was judged that modeling the entire primary to secondary leakage as entering the intact SGs is conservative.

It is noted that the approach of modeling the primary to secondary leakage as entering the intact SGs has been previously accepted by the NRC for Diablo Canyon (ADAMS Accession No. ML17012A246, Section 3.3.7).

RAI ARCB1-SGTR-4 - SGTR

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

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

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

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

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

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

The LAR proposes to change the existing bases as follows:

> The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at or more conservative than the LCO limits, and ~~an existing~~ reactor coolant steam generator (SG) tube leakage rate of 1 gpm exists or results from accident induced conditions.

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

Response

The dose analyses model 1 gpm of primary-to-secondary leakage. Whether the leakage is pre-existing or accident-induced, the leakage is equivalent for the purposes of calculating offsite and control room doses. It is noted that some quantity of pre-existing leakage is assumed as there is activity in the secondary side prior to the start of the event. However, the quantity of activity is based upon the allowable technical specification concentration, not on a specific leakage rate.

Based on the NRC's observation, Enclosure IV, Section 4.3.8.2.2 should be revised to remove the words "accident-induced." The revised sentence is: "The entire 1 gpm primary-to-secondary leakage allowed by the TS is assumed to be leaking into the intact SGs [steam generators] with a density based on cooled liquid, which otherwise is negligible compared to the flow through the ruptured tube.".

<u>RAI ARCB1-SGTR-5</u> - SGTR, MSLB, and other accidents that assume DEX-133

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

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

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

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

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

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

WCNOC has voluntarily proposed to revise the design basis radiological analysis under 10 CFR 50.67. In the proposed design bases analysis the condition allowed in the WCGS TSs, which allows continued operation with the DEX-133 at levels higher than 500 micro-Curies per gram, is an unanalyzed condition where compliance with 10 CFR 50.67 is unknown. Note that the current and proposed TS 3.4.16 Bases states that these analyses <u>assumed values equal to or greater than 500 micro-Curies per gram DEX-133</u>.

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

CFR 50.67 are met. In order for the staff to make its assessment of reasonable assurance please provide the following information:

1. For every accident that assumes the RCS activity is based upon the value of the DEX 133 specified in TS 3.4.16, "RCS Specific Activity" please submit for the NRC staff's review a revised radiological consequences analyses that assumes the DEX 133, allowed by the proposed TSs (values equal to or greater than 500 micro-Ci/gm) at the start of the event and show that the dose results meet the limits in GDC 19 of 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).

Response

The dose analyses model noble gases initially in the primary coolant at a concentration of 500 μCi/gm dose equivalent Xe-133. This represents an upper limit on allowable noble gas coolant concentration. The Technical Specification Bases for LCO 3.4.16 will be revised to remove the words "or greater than." The revised sentence is:

"In both cases, the noble gas specific activity is assumed to be equal to 500 μCi/gm [micro-Curies per gram] DOSE EQUIVALENT XE-133."

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

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

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

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

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

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

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

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

> A loss of offsite power was assumed.

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

Response

The loss of offsite power is taken to be a consequence of the reactor trip generated by the initiating event, not a second, independent event occurring simultaneously. The analysis performed for the Locked Rotor event shows that the reactor coolant low-flow reactor trip setpoint is reached almost immediately after event initiation, followed by a reactor trip signal one second later. For conservatism, the resulting loss of offsite power is modeled to occur coincident with the reactor trip, as this further reduces reactor coolant flow through the core. The locked rotor rods-in-DNB analysis is described in Section 2.4.2 of Enclosure I of "Wolf Creek, License Amendment Request for the Transition to Westinghouse Core Design and Safety Analyses" (ADAMS Accession No. ML17054C103). As shown in Table 2.4.2-1 of Enclosure I, the analysis results show a reactor trip and subsequent loss of offsite power occurring in less than 2 seconds and minimum DNBR occurring at 3.2 seconds.

As a modeling simplification for the dose analysis, the activity released from the damaged fuel is modeled as being available for release from the RCS at the start of the event and the releases from the SGs to the environment are modeled as beginning at the start of the event.

<u>RAI ARCB1-MSLB-1</u> - Main Steam Line Break (MSLB)

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

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

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

> The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

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

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

Response

A loss of offsite power was assumed in the main steamline break dose analysis to maximize the postulated MSLB radiological consequences. The loss of offsite power is taken to be a consequence of the reactor trip generated by the initiating event, not a second, independent event occurring simultaneously. Following a main steamline break, the reactor trip and resulting loss of offsite power would be expected within a few seconds. As a modeling simplification, releases from the intact SGs to the environment are modeled as beginning at the start of the event.

RAI ARCB1-MSLB-2 - MSLB

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

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

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

> During startup or shutdown evolutions, when the operator manually blocks the safety injection on low pressurizer pressure or low steamline pressure and steamline isolation on low steamline pressure when pressurizer pressure is less than P-11 setpoint (i.e., 1970 psig), the steamline pressure-negative rate-high signal is automatically enabled to provide steamline isolation. For inside containment breaks, steamline isolation may also be provided by the containment pressure High-2 signal and safety injection would be actuated by the containment pressure High-1 signal. For a steamline break occurring outside containment, an automatic actuation signal for safety injection would not be available.
>
> Note that the conclusion that the hot-zero power case is the limiting case is based on certain specific protection system performance characteristics credited for "at power" steamline break analyses.

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

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

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

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

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

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

Response

In response to this RAI, the Main Steamline Break control room dose analysis was revised and a conservative time to isolate the control room was chosen. A conservative time of 2 minutes, concurrent with the completion of the faulted SG blowdown was selected. Relative to the completion of the faulted SG blowdown, earlier control room isolation reduces the activity ingress due to filtration and later control room isolation reduces the activity via purging of the control room. Thus, control room isolation coincident with the completion of the faulted SG blowdown is conservative. No other input or assumptions were revised.

The revised MSLB control room doses are 4.8 rem TEDE for the accident-initiated iodine spike and 4.5 rem TEDE for the pre-accident iodine spike.

It is recognized that below the P-11 interlock, manual SI actuation would be required to isolate the control room. It is expected that this action could be accomplished within a reasonable amount of time (i.e. <30 minutes). Sensitivity analyses examining the effects of isolating the control room at 30 minutes past the start of the event yield doses approximately 30% lower than those resulting from isolation coincident with the end of faulted SG blowdown. The dose reduction is driven by the purging of the faulted SG blowdown activity from the control room prior to isolation. It is noted that the dose contributions from primary to secondary leakage increased, although the increase was more than offset by the decrease in the faulted SG blowdown contribution. It is anticipated that the primary to secondary leakage would eventually come to dominate the dose and could result in unacceptable doses. The time this is expected to occur is after two hours. Thus, while control room isolation following a MSLB occurring when the P-11 interlock allows blocking the automatic SI signal is required, the action is not a time critical action.

This change to the analysis would have the following effect on Enclosure IV of the LAR.

Section 4.3.3.2.3 would be revised to read as follows:

4.3.3.2.3 Control Room

In the event of an MSLB, the low steamline pressure SI setpoint will be reached almost immediately following the break. The SI signal causes the control room to switch from the normal operation mode to the emergency operation mode. The switchover is conservatively modeled at 2 minutes, concurrent with the completion of the faulted SG blowdown. Relative to the completion of the faulted SG blowdown, earlier control room isolation reduces the activity ingress due to filtration and later control room isolation reduces the activity via purging of the control room. Thus, control room isolation coincident with the completion of the faulted SG blowdown is conservative. As discussed in Section 4.3.2.1, operator action is taken 90 minutes after event initiation to isolate the ventilation train with failed filtration.

It is recognized that below the P-11 interlock, manual SI actuation would be required to isolate the control room. It is expected that this action could be accomplished within a reasonable amount of time (i.e. <30 minutes). Sensitivity analyses examining the effects of isolating the control room at 30 minutes past the start of the event yield doses approximately 30% lower than those resulting from isolation coincident with the end of faulted SG blowdown. The dose reduction is driven by the purging of the faulted SG blowdown activity from the control room prior to isolation. It is noted that the dose contributions from primary to secondary leakage increased, although the increase was more than offset by the decrease in the faulted SG blowdown contribution. It is anticipated that the primary to secondary leakage would eventually come to dominate the dose and could result in unacceptable doses. The time this is expected to occur is after two hours. Thus, while control room isolation following a MSLB occurring when the P-11 interlock allows blocking the automatic SI signal is required, the action is not a time critical action.

4.3.3.4 Results and Conclusions

The following MSLB accident doses in Section 4.3.3.4 would be modified:

For the pre-accident iodine spike case:

- Control room 4.5 rem TEDE

For the accident-initiated iodine spike case:

- Control room 4.8 rem TEDE

Table 4.3-6 would be impacted to reflect the new control room switchover assumption:

Table 4.3-6 Assumptions Used for Main Steamline Break Analysis

- Delete row "Safety injection (SI) signal (sec)"
- Change row "Control room isolation (including delay) (sec)" to "Control room isolation (min)" and change the associated AST value to 2.

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

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

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

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

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

Response

The nuclides Kr-83m, Kr-89, and Xe-137 were excluded from the AST tank rupture calculations because they do not contribute to the dose. From Table III.1 of Environmental Protection Agency Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA-402-R-93-081, September 1993, Kr-83m has a dose conversion factor ("Effective") of 1.50E-18 Sv-m3/Bq-sec; this is approximately 2 orders of magnitude lower than the Kr-85 dose conversion factor ("Effective") of 1.19E-16 Sv-m3/Bq-sec, which is the next lowest Krypton dose conversion factor. From Enclosure IV, Table 4.2-4, Kr-83m has a calculated inventory of 1.92E+01 Ci; this is approximately 2 orders of magnitude lower than the Kr-85 inventory of 5.52E+03 Ci. With 2 orders of magnitude less inventory than Kr-85 and 2 orders of magnitude less dose impact than Kr-85, Kr-83m is judged to not contribute to the calculated dose and so is not modeled.

Dose conversion factors for Kr-89 and Xe-137 are not listed in Table III.1 of Environmental Protection Agency Federal Guidance Report No. 12, and so these nuclides are assumed not to contribute to the dose.

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

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

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

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

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

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

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

Response:

Control room filters were not credited in the AST tank rupture calculations. The cited text from Enclosure IV, Section 4.3.10.2.1, "Source Term," is revised as follows:

> The iodine is assumed to be 100% elemental; however, the chemical species of iodine has no impact on the calculation since no removal processes are modeled.

The entirety of the waste gas tank inventory is assumed to be released linearly over a two hour period, as stated in Enclosure IV, Section 4.3.10.2.2. There are no removal processes (e.g. filtration or plateout) modeled in the release of the waste gas decay tank inventory. There is no partitioning modeled in the release of the waste gas decay tank inventory. Control room filtration is not modeled as a result of release of the waste gas decay tank inventory. Dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," do not distinguish between iodine chemical species. Partitioning of the iodine in the VCT is conservatively addressed as explained in the response to ARCB1-WT-4 by using a total partition coefficient of 100 for the RCS chemical species split of 95% particulate, 4.85% elemental, and 0.15% organic. Since no further partitioning or filtered removal is applied to the source term in the gas space, and since there is identical dose impact from the different iodine chemical species, considering 100% elemental iodine has no impact on the dose results.

<u>Request for Additional Information ARCB1-WT-4</u> - Waste Gas Decay Tank

Section 4.2, "Accident Source Terms," of Enclosure IV to the letter dated January 17, 2017 states:

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

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

Response

The conservative assumption for the hypothetical liquid waste tank inventory is that the input to each of the tanks is based on continuous full power operation with [ ]^{a,c}. Based on this activity concentration, it is assumed that each of the feed tanks is filled and drained such that [ ]^{a,c}. Thus, the feed rate into a tank must equal the discharge rate from the tank into the hypothetical tank. Also, the (equilibrium) activity concentration in the feed tank comprises [ ]^{a,c} where equilibrium activity levels are assumed to exist (based on decay alone).

Consider the differential equation defining the activity in any of the feed tanks,

[ ]^{a,c}

where:

N_T = concentration of nuclide in the tank, atoms/gram
N_{RCS} = concentration of nuclide in the water, atoms/gram
F_{PCA} = [ ]^{a,c}
Q = flow rate into a tank, grams/day
M = mass of water in the tank, grams
DF = decontamination factor = [ ]^{a,c}
λ = nuclide decay constant, 1/day

Let,

γ = production rate = [ ]^{a,c}

and

β = removal rate = [ ]^{a,c}

Then, multiplying through by the integrating factor, [ ]^{a,c}, and rearranging,

[ ]^{a,c}

Then, since,

[ ]^{a,c}

then

[ ]^{a,c}

Taking the integral of both sides

[ ]^{a,c}

gives,

[ ]^{a,c}

or

[ ]^{a,c}

In the limit, $N_T = 0$ when $t = 0$; then,

[ ]^{a,c}

and

[ ]^{a,c}

Then,

[ ]^{a,c}

Now, replacing the γ and β quantities,

[ ]^{a,c}

At equilibrium, the exponential quantity equals one, and the equation defining the activity inventory in the feed tank, I_T, is obtained by [

]^{a,c}, i.e.

[ ]^{a,c}

or

[ ]^{a,c}

The activity concentration in the tank is [

]^{a,c} or,

[ ]^{a,c}

The rate of activity entering the hypothetical waste tank is this concentration multiplied by [

]^{a,c},

[ ]^{a,c}

and at equilibrium, with only decay as the removal mechanism, the inventory in the hypothetical waste tank is:

[ ]^{a,c}

Request for Additional Information ARCB1-WT-5 - Waste Gas Decay Tank

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

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

In Section 8, "NRC Regulatory Issue Summary 2006-04 Comparison," of Enclosure IV to the letter dated January 17, 2017 WCNOC states the submittal conforms to this RIS position.

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

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

And

> In addition to the noble gases, radioactive iodine is considered to partition in the VCT vapor space (a partition factor = 100) and to be carried over into the gas decay tank during the degassing operations. [1% of the iodine is released]

And

> Two separate scenarios for the accidental release of activity from liquid waste tanks are analyzed; that is, 1) the failure of a recycle holdup tank from which the release of 100% of the noble gas nuclides and 10% of the iodine activity in the tank is assumed, and 2) the failure of a hypothetical liquid waste tank from which 100% of the iodine activity is released.

Enclosure IV to the letter dated January 17, 2017 provided two tables, Table 4.3-13 and Table 4.3-14, that states that the iodine chemical form modeled in the Waste Gas Decay Tank Failure analysis and the

Liquid Waste Tank Failure analysis assumes that all the iodine in these tanks is assumed to be 100% elemental iodine and this is a change from the current licensing basis. .

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

2) Several different release assumptions for the iodine activity are assumed above. Please justify the partition factor of 1 percent and provide enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. Some factors that should be considered 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.

Responses

1) The two potential failures of the gas tanks are the failure of the shutdown operation gas tanks or the failure of the normal operation gas tanks. The source term considered for the waste gas decay tank failure consisted of the gaseous inventory in the primary coolant system that is removed during shutdown operations. This activity is directed to one of the two "shutdown' gas decay tanks in the WCGS design. At this point, the activity is allowed to decay whereby all activity, with the exception of Kr-85, is essentially eliminated due to the relatively short half-lives of other gaseous radionuclides. After the 60-day period, the remaining activity (consisting of largely Kr-85 with trace amounts of other nuclides) is transferred to one or more of the six "normal operation" tanks. There is not a 2 cycle restriction imposed on the "normal operation" tanks; however, since a "shutdown" gas decay tank would yield more limiting results due to the relatively high amount of activity from nuclides with shorter half-lives, it was used rather than a "normal operation" tank with a 40-year buildup of Kr-85. As such, it would be overly conservative to assume that a combination of the shutdown gas activity and a 40-year buildup of Kr-85 are present in the same tank, given that the 40-year Kr-85 activity would be accumulating in one of the "normal operation" gas decay tanks (which are isolated from the "shutdown" gas decay tanks during the shutdown degassing).

2) The partition factor of 100 applied to the iodine in the volume control tank is based on the recommended total iodine separation factor from WCAP-8159. The factors compiled in WCAP-8159 are the result of conservative consideration of in-plant measurements and laboratory mock-ups. Since this partition factor is for total iodine, the specific species of iodine is not considered in the calculation of activity in the volume control tank. Note that as previously stated in Sections 4.3.10 and 4.3.11 of the Alternative Source Term Transmittal (Enclosure IV of the submittal), the iodine species assumed for the dose calculation of the Waste Gas Decay and Volume Control tanks is all elemental since no chemical species-dependent removal mechanisms (e.g., control room filters) are credited.

   In addition, the following aspects are considered in justifying the partition factor of 100 for the total iodine in the volume control tank:

- The calculation of volume control tank iodine concentration conservatively neglects the removal of iodine activity via the reactor coolant filter which would remove a significant portion of the iodine in the system. From Regulatory Guide 1.183, the chemical composition of radioiodine released from the reactor coolant system is 95% particulate, 4.85% elemental, and 0.15% organic, suggesting that as much as 95% of the RCS iodine is subject to removal by filters.

- The volume control tank is typically maintained at subcooled conditions, which would result in only elemental and organic iodines becoming airborne and transferred to the waste gas decay tank.

- From Section 3.3 of NUREG/CR-5950, the partition coefficient for elemental iodine can be calculations by the following calculation:

  $$\text{Log}_{10}\, PC(I_2) = 6.29 - 0.0149T$$

  where T is the temperature in Kelvin

  The typical temperature of the volume control tank is 110°F (316.5 K). Therefore, the partition coefficient of the elemental iodine in the VCT is approximately 38. Given that particulate iodine remains in the liquid and the partition coefficient for organic iodine is 1, the total release of iodine into the volume control tank gas space for the Regulatory Guide 1.183 chemical form of iodine in the RCS is less than 0.3% of the total iodine, which is less than the overall partition coefficient for iodine of 100 (equating to 1% release to the gas space).

- The conversion of particulate iodine to elemental iodine is neglected due to the pH of the volume control tank and the relatively low concentration of total iodine in the solution. From the WCGS USAR Table 5.2-5, the recommended reactor coolant system water chemistry is to maintain solution pH between 4.2 and 10.5. From the table below, the total iodine concentration in the volume control tank is calculated from the volume control tank activity concentrations, nuclide-specific half- lives, and mass of water assumed in the volume control tank (4.53E6 g). From the calculations, the total iodine concentration (assuming a conservative molar mass of 127 and water volume of 160 ft^3) is approximately 5E-10 g-atoms/L, with the majority being I-127 and I-129, which are not dose significant.

Nuclide	Volume Control Tank Activity Concentration (µCi/g)	Half Life	Nuclide Mass Concentration (g/g water)	Nuclide Mass Inventory (g)
I-127	-	Stable	1.24E-11	5.62E-05
I-129	7.17E-09	1.57E7 y	4.06E-11	1.84E-04
I-130	4.65E-03	12.36 h	2.38E-15	1.08E-08
I-131	3.28E-01	8.02 d	2.64E-12	1.20E-05
I-132	3.39E-01	2.28 h	3.26E-14	1.47E-07
I-133	5.04E-01	20.8 h	4.45E-13	2.02E-06
I-134	7.30E-02	52.6 m	2.74E-15	1.24E-08
I-135	2.85E-01	6.57 h	8.07E-14	3.65E-07
			Total	2.55E-04

Figure 3.1 of NUREG/CR-5950 provides curves of fractional elemental iodine based on pH and total iodine concentration in a system. The figure only contains curves for iodine concentrations as low as 10^{-6} g-atoms/L. At a pH of 4.2 and iodine concentration of 10^{-6} g-atoms/L, the fraction of total iodine as elemental iodine is less than 0.05. Given the actual iodine concentration for dose significant nuclides in the volume control tank is several orders of magnitude lower, it can be conservatively assumed that the fraction of conversion to elemental iodine is significantly lower than the curve for a concentration of 10^{-6} g-atoms/L, which is a negligible amount.

The assumption of 10% release for iodine is based on a decontamination factor of 10 attributed to the mixed bed demineralizer upstream of the volume control tank. The decontamination factor of 10 for iodine is a conservative assumption that is consistent with the current analysis of record for WCGS. From NUREG-0017, Revision 1, a mixed bed demineralizer decontamination factor for the primary coolant letdown system is at least 50, for all nuclides with the exception of alkali metals cesium and rubidium. Therefore, the 10% release for iodines is justified.

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

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

> The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

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

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

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

1. Please state whether the loss of offsite power was assumed to maximize the postulated Control Rod Ejection Accident radiological consequences. If a methodology other than that in RG 1.183 is used please provide details about the methodology and justify the proposed change from the current methodology and why it is conservative.

Response

A loss of offsite power was assumed for the control rod ejection dose analysis to maximize the postulated Control Rod Ejection Accident radiological consequences. The loss of offsite power is taken to be a consequence of the reactor trip generated by the initiating event, not a second, independent event occurring simultaneously. Following a control rod ejection, the reactor trip and resulting loss of offsite power would be expected within a few seconds. As a modeling simplification, releases to the environment via the secondary relief valves are modeled as beginning at the start of the event.

<u>RAI ARCB1-CREA-2</u> Control Rod Ejection Accident - CREA

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

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

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

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

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

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

Response

1. Explanations of how the Regulatory Positions 5.5 and 5.6 were considered in the analysis are listed below.

5.5.1 There is no period of steam generator dryout in the rod ejection analysis. The steam generator tubes remain submerged, and thus, the primary to secondary leakage is assumed to mix with the bulk water in the steam generator without flashing. This is consistent with Position 5.5.1.

5.5.2 As there is no flashing of primary to secondary leakage, the scrubbing models referred to in Position 5.5.2 are not applicable.

5.5.3 The leakage does not flash and is assumed to mix with the bulk water. This is consistent with Position 5.5.3.

5.5.4 The radioactivity in the water is assumed to become airborne as a function of the steaming rate and a partition coefficient. A partition factor of 100 is applied to the iodines for these steaming

releases. The moisture carryover of 0.25% is applied to the alkali metals for these steaming releases. This is consistent with Position 5.5.4.

5.6 The issue of tube uncovery was addressed by the Westinghouse Owners Group (WOG) in WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992. The WOG program concluded that the effect of tube uncovery would be essentially negligible and the issue could be closed without any further investigation or generic restrictions. This position was accepted by the NRC in a letter dated March 10, 1993, from Robert C. Jones, Chief of the Reactor System Branch, to Lawrence A. Walsh, Chairman of the WOG. The letter states "... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue to be resolved." Consistent with this position, the rod ejection dose analysis did not model tube uncovery in the steam generators. This addresses the concerns raised in Position 5.6.

2. The partition coefficient of 100 is applied only to the radioactivity in the bulk water assumed to become vapor by steaming. No primary-to-secondary leakage is assumed to immediately flash and rise through the bulk water as discussed above related to Regulatory Position 5.5.1.

RAI ARCB1-CONTROL ROOM-1 - All DBAs

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

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

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

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

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

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

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

Response

The shine dose contribution to the post-LOCA control room dose from the activity in the control building was calculated using control building radionuclide inventories taken from the RADTRAD output. Activity was assumed to be distributed uniformly throughout the control building, excluding the control room. Dose calculations were performed using a SCAP-II model.

The SCAP-II model represents the various rooms in the control building above and below the control room. The walls, floors, and ceilings were modeled using concrete as the shielding source between the control room and other control building rooms. The dimensions of the various rooms and thicknesses of the walls, floors, and ceilings were based on plant-specific drawings.

The post-LOCA 30-day integrated dose to the control room operators from activity dispersed within the control building was calculated to be approximately 29 mrem (maximum). The 29 mrem contribution from activity in the control building is approximately 0.6% of the overall dose of 4.8 rem reported for LOCA (not including the dose to the operator in transit to and from the parking lot). Thus, it was concluded that the activity remaining in the control building is not a significant dose contributor. Note that the dose of 29 mrem conservatively was not recalculated in the updates provided in the response to ARCB1-CONTROL-ROOM-3 which reduced the RWST back-leakage rate from 3.8 gpm to 2.0 gpm.

Thus, the contribution to control room doses for events other than LOCA was considered and judged to be within the rounding applied given the conservative approximation of 1% of the total dose from control building activity.

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

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

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

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

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

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

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

Response

A maximum RCS mass is used in the determination the accident-initiated iodine spike appearance rates. In that calculation, a maximum RCS mass results in a higher equilibrium appearance rate. This then results in conservative spike appearance rates. The maximum RCS mass is only used for the equilibrium appearance rate calculation. It is not modeled as the RCS mass in other parts of the dose analyses for reasons stated below.

A lower RCS mass will maximize the concentration of activity in the RCS when the activity released from the fuel is modeled, either from failed fuel or from an accident initiated iodine spike.

For modeling the initial primary coolant activity, the initial activity in the RCS is defined by the Technical Specification limits on the primary coolant concentrations and is not impacted by the assumed mass.

From Table 4.3-5 of Enclosure IV, the radiation detector setpoint is 2.12E-03 μCi/cc Xe-133. The Xe-133 concentration of the primary coolant is determined either by Technical Specifications (and so is not impacted by the assumed mass) and by the quantity, if any, released from the failed fuel divided by the RCS mass (for which a lower RCS mass is conservative).

Request for Additional Information ARCB1-CONTROL ROOM-3 - Control Room

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

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

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

Westinghouse Response

In order to accommodate the increase in control room dose from operator transit to and from the control room, conservatism in the analyzed value for the RWST back-leakage was removed in order to lower the overall dose. The RWST back-leakage was reduced from 3.8 gpm to 2.0 gpm. The LOCA dose was re-analyzed with the lower leakage and the following dose results were obtained, to replace the RWST back-leakage results from Section 4.3.9 of Enclosure IV:

EAB (0.5 hr to 2.5 hr):

- Containment Leakage 4.73 rem TEDE
- ECCS Leakage 0.463 rem TEDE
- RWST Back-Leakage 0.00118 rem TEDE
- Containment Purge 0.0 rem TEDE

LPZ:

- Containment Leakage 2.00 rem TEDE
- ECCS Leakage 1.26 rem TEDE
- RWST Back-Leakage 0.374 rem TEDE
- Containment Purge 0.000824 rem TEDE

Control Room:

- Containment Leakage 1.01 rem TEDE
- ECCS Leakage 0.848 rem TEDE
- RWST Back-Leakage 0.632 rem TEDE
- Containment Purge 0.153 rem TEDE

The total updated LOCA doses from all the above pathways are listed below:

- EAB 5.2 rem TEDE
- LPZ 3.7 rem TEDE
- Control Room 2.8 rem TEDE
- TSC 4.3 rem TEDE

The operator dose from ingress and egress to the control room during the entire 30-day LOCA event has been evaluated. The assumption inherent in the transit dose is that the operator travels to the site, parks in the parking lot, and walks from the parking lot to the control building via the turbine building entrance without any supplemental radiation protection. The operator leaves using the same path. The transit dose calculation used the previously defined LOCA dose evaluation models and numerous other inputs and assumptions, detailed below. The scenario is not the most likely scenario, as it ignores the principal of ALARA, but the modeling and assumptions used are consistent with a design basis accident.

The dose to the operator was evaluated to be 0.59 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. Thus, the total control room dose (including dose to the operator inside the control room) is 3.4 rem TEDE.

Control Room:

- Containment Leakage 1.01 rem TEDE
- ECCS Leakage 0.848 rem TEDE
- RWST Back-Leakage 0.632 rem TEDE
- Containment Purge 0.153 rem TEDE
- External Sources 0.59 rem TEDE
- Total 3.4 rem TEDE

The RADTRAD code is used to calculate both the dose from airborne activity and the dose from deposited activity. The airborne activity dose is calculated using the non-control room dose locations to appropriately calculate the inhalation and immersion dose from activity released to the environment at the various locations along the operator transit route. The deposited activity dose is calculated using the control room dose location as this is the only RADTRAD dose location which will allow for the accumulation and retention of activity which would be consistent with ground-deposited activity. The control room dose location is modeled for various locations along the operator transit route.

Design Inputs/Assumptions:

1. Meteorological data is the same as used for the LOCA dose consequences analysis previously discussed in the submittal.
2. Containment leakage, quantity and timing, is the same as modelled in the LOCA dose consequences analysis.
3. ESF leakage outside containment, quantity and timing, is the same as modelled in the LOCA dose consequences analysis.
4. RWST leakage, quantity and timings, is the same as calculated in the LOCA dose consequences calculation (when accounting for the reduction in RWST back-leakage to 2.0 gpm).
5. Containment diameter is modeled as a cylindrical component up to 136.5 feet in elevation with a diameter of 70 feet, 0.25-inch steel plate liner, and exterior concrete thickness of 4 feet and a hemispherical dome from 136.5 feet to 209.52 feet with a 0.25-inch steel plate liner, and exterior concrete thickness of 3 feet. This is the same as modelled in the LOCA dose consequences analysis.

6. Walking distance from the parking lot to the entrance to the turbine building is conservatively assumed to be 1025 feet. This was determined based on a typical walking path of the operator from the parking lot to the entrance to the turbine building.
7. Operator walking speed is 3 miles per hour (mph). The resulting transit time is conservatively modeled as 4.1 minutes.
8. The operators will be confined to the control room for the first 24 hours (100% occupancy factor) following the accident. The first ingress/egress will begin at 24 hours following the event and be repeated every 12 hours for the length of the accident (30 days, 59 trips total).
9. Operator respiration is the respiration rate for the control room (3.5E-04 m^3/sec) as defined by regulatory guidance.
10. For the ground shine dose, deposition velocities of 1.0E-02 m/sec for elemental iodine, 1.0E-04 m/sec for organic iodine, and 1.0E-03 m/sec for particulates are modeled, consistent with NUREG/CR-3332. Noble gases are not assumed to deposit on the ground.
11. The dose point for the operator walking along the ingress/egress path is 5 feet off the ground.
12. Shielding provided by intervening equipment, plant buildings, or other materials is ignored (except for the containment concrete and liner when determining the containment shine dose).
13. Dose conversion factors for airborne activity are the same as used for the LOCA dose consequences analysis previously discussed in the submittal. The dose conversion factors for deposited activity are taken from Federal Guidance Report No. 13 (Electronic Supplement).
14. Six points along the ingress/egress path are used to determine dose rates (See Table 1 and Figure 1 for the location of these points). Atmospheric dispersion factors (X/Qs) are calculated for these dose points and are given in Table 2. A weighted X/Q for the entire length of the pathway is modeled for each transit based on the transit segment distance. For example, the operator is modeled to be at Point 1 while traveling the 200 feet to reach Point 2. This results in a conservative X/Q for the transit since no benefit is taken for the lower X/Q values which would result during intermediate locations in the transit segments which are further from the release points.

Table 1: Operator Transit Segments

Transit Segment	Starting Location	End Location	Segment Distance (ft)
1	Point 1	Point 2	200
2	Point 2	Point 3	175
3	Point 3	Point 4	125
4	Point 4	Point 5	200
5	Point 5	Point 6	125
6	Point 6	Parking Lot	200

Table 2: Dose Point X/Qs

Time	Point 1	Point 2	Point 3	Point 4	Point 5	Point 6
Containment Leakage (Maximum of Unit Vent, Equipment Hatch, and Reactor Building Wall)						
0-2 hours	1.80E-03	9.36E-04	1.61E-03	8.96E-04	6.67E-04	4.89E-04
2-8 hours	1.49E-03	6.28E-04	8.00E-04	4.78E-04	4.72E-04	3.79E-04
8-24 hours	5.73E-04	2.30E-04	3.28E-04	1.97E-04	1.90E-04	1.53E-04
24-96 hours	4.90E-04	1.68E-04	2.07E-04	1.31E-04	1.23E-04	9.57E-05
96-720 hours	3.62E-04	1.29E-04	1.51E-04	9.32E-05	9.67E-05	7.76E-05
ECCS Leakage (Unit Vent)						
0-2 hours	8.94E-04	4.96E-04	5.56E-04	4.17E-04	3.27E-04	2.67E-04
2-8 hours	6.86E-04	3.02E-04	3.15E-04	2.43E-04	2.15E-04	1.86E-04
8-24 hours	2.68E-04	1.10E-04	1.25E-04	9.65E-05	8.82E-05	7.39E-05
24-96 hours	2.11E-04	7.03E-05	7.59E-05	5.64E-05	5.63E-05	4.92E-05
96-720 hours	1.74E-04	5.08E-05	6.28E-05	4.93E-05	4.57E-05	3.95E-05
RWST Back-Leakage (RWST)						
0-2 hours	5.15E-04	3.16E-04	4.48E-04	3.48E-04	3.59E-04	3.56E-04
2-8 hours	4.05E-04	2.10E-04	2.52E-04	1.75E-04	1.95E-04	2.28E-04
8-24 hours	1.55E-04	7.71E-05	9.04E-05	6.88E-05	8.03E-05	8.77E-05
24-96 hours	1.24E-04	5.65E-05	6.09E-05	4.30E-05	5.38E-05	6.08E-05
96-720 hours	9.25E-05	4.37E-05	4.67E-05	3.51E-05	3.79E-05	4.38E-05

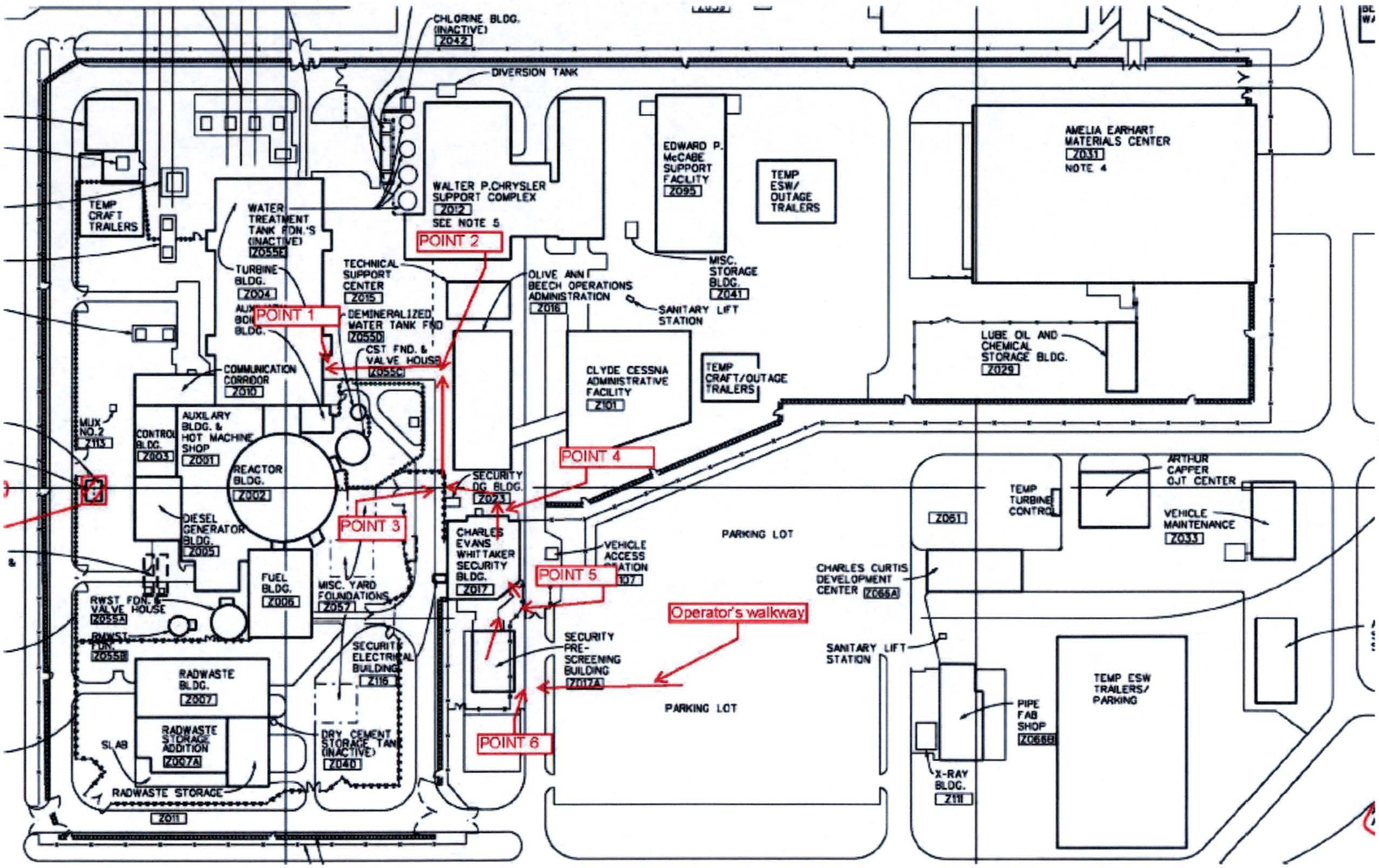

Figure 1: Dose Point Locations and Operator Transit Pathway

<u>Request for Additional Information ARCB1-CONTROL ROOM-5</u> - Control Room and TSC

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

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

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

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

Response

Table 4.2-1 of Enclosure IV presents the results of the ORIGEN-S calculation performed whereas Table 4.3-1a of Enclosure IV presents the subset of the ORIGEN-S inventories used in the dose calculations. These nuclides were selected based on expected dose significance. The only halogens modeled in the dose analyses are the iodines, as presented in Table 4.3-1a of Enclosure IV. Bromines are not modeled in the analyses because of their low dose significance. Table 4.3-12 of Enclosure IV is consistent with Table 4.3-1a of Enclosure IV.

The low expected dose significance of the bromines is based on the low inventories and low inhalation dose conversion factors of the bromines relative to the iodines. From Table 4.2-1 of Enclosure IV, the bromine inventories in the core are an order of magnitude lower than the iodines (except I-130). Similar to iodines, the dominant component of the bromine TEDE dose is inhalation. Inhalation dose conversion factors are taken from Table 2.1 of Environmental Protection Agency Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, Environmental Protection Agency, September 1988. Br-83 and Br-84 have inhalation dose conversion factors of 2.41E-11 Sv/Bq and 2.61E-11 Sv/Bq, respectively. These are lower than any of the iodine listed in Table 4.3-1a of Enclosure IV, including I-134. Note that Br-85 is not listed in the Federal Guidance Report, and so is assumed to have no dose significance. The lower inventories combined with the lower inhalation dose conversion factors yield low dose significance, and thus, the bromines were not modeled in the analysis.

RAI ARCB1-GENERAL-1 - Dose Conversion Factors

RG 1.183, Regulatory Position 4.1.4 states:

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

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

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

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

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

Response

The following statement should be added to Table 4.3-3a of Enclosure IV as a footnote:

Cs-137 has a very low EDE DCF value (7.74E-18 Sv-m^3/Bq-sec). However, its short-lived daughter nuclide Ba-137m has a large EDE DCF (2.88E-14 Sv-m^3/Bq-sec). The daughter nuclide value was substituted for the parent nuclide value in the analyses.

Request for Additional Information ARCB1-GENERAL-3 - All DBAs

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

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

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

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

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

Response

The doses are calculated using the inputs, methods and assumptions identified in Section 4.3 of Enclosure IV, as clarified by the Requests for Additional Information. Upon completion of the dose calculations, additional conservatism was added to the calculated dose values to arrive at the reported values. The purpose in this additional conservatism is to provide additional margin, should a small error or non-conservatism be identified and changes made pursuant to 10 CFR 50.59. This additional conservatism was added by increasing the calculated doses by approximately 10% (unless that would cause a limit to be exceeded) and then rounding up to 2 significant figures. Table 1 summarizes the increases applied prior to rounding.

Table 1: Increases to Calculated Doses

Event/Location	EAB	LPZ	CR	TSC
MSLB	10%	10%	10%	10%
LOAC	10%	10%	10%	10%
LR	10%	10%	10%	10%
CREA	10%	10%	10%	10%
LLB	10%	10%	10%	10%
SGTR	10%	10%	10%	10%
LOCA	10%	10%	5%	0%
Tank Ruptures	10%	10%	10%	10%
FHA	10%	10%	10%	10%

The calculation notes and RADTRAD input and output files supporting Enclosure IV and the revised analyses produced in response to RAIs will be made available upon request.