Responses to NRC RAIs on August 2013 Methodology Transition LAR Submittal (Non-Proprietary)

November 2016

WCNOC previously submitted a license amendment request (LAR) on August 13, 2013 (Reference 1) requesting approval of changes to the Wolf Creek Generating Station (WCGS) Technical Specifications (TSs). That application proposed the transition to Westinghouse core design and safety analysis methodologies, full scope implementation of Alternative Source Term (AST), and implementation of instrumentation setpoint and control uncertainty calculations based on the current Westinghouse Setpoint Methodology (including adoption of Option A of Technical Specification Task Force (TSTF) TSTF-493-A, Revision 4). WCNOC subsequently withdrew the LAR on June 18, 2014 (Reference 3) based on deficiencies discovered by WCNOC.

References 2, 4, 24 and 25 are Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAI) related to Reference 1. WCNOC letters ET 14-0003 (Reference 20) and WO 14-0032 (Reference 22) provided responses to the RAIs received in Reference 24. WCNOC letter WO 14-0031 (Reference 21) provided responses to the RAIs received in Reference 25. WCNOC withdrew the LAR prior to providing a response to References 2 and 4. WCNOC has subsequently developed responses to the RAIs received in References 2 and 4. The responses are provided within this enclosure and the applicable information has been incorporated into the current LAR.

Following the withdrawal of the LAR, two changes were made that affected the responses contained within References 20, 21, and 22. First, WCGS implemented the AST sump pH analysis (refer to Section 4.4 of Enclosure IV) as its licensing basis analysis under the 50.59 process. Second, it was determined that a modification would be implemented to support full implementation of AST. Specifically, the operator action to isolate the failed CREVS train (Discussed in Section 15A.3.1 of the USAR) has historically required local operator action. However, in order to reduce the difficulty of completing this time critical action and to eliminate dose exposure to a local operator, a design modification will be implemented to supply the CREVS control room isolation dampers with Class IE 120 VAC battery power. This modification will be implemented under the 50.59 process. Based upon these two changes, responses previously transmitted in References 20, 21, and 22 have been updated and are included within this enclosure (in addition to the unchanged responses previously provided). Specifically, the changes to the affected responses (EMCB-RAI-1, ESGB-RAI-1, and EEEB-RAI-2) are shown in bold.

The responses to the RAIs are applicable to the current LAR and are therefore included in this Enclosure. Enclosure VIII provides supplemental documents that were requested by ARCB-RAI-33 and ARCB-RAI-36. The specific NRC request is provided in italics.

Note that Reference 27 also provided NRC RAIs related to Reference 1. These RAIs are associated with the original proposed addition of TSTF-493 and, as such, no longer require a response.

Reference 24 RAIs as clarified by Reference 26

EMCB-RAI-1

Implementation of an Alternate Source Term (AST) in accordance with Section 50.67, “Accident source term,” of Title 10 of the Code of Federal Regulations (10 CFR) could affect structures, systems, and components (SSCs) which were not previously evaluated for the consequences of a design-basis accident (DBA) and, as such, may not be seismically qualified. Appendix A to 10 CFR Part 100, “Reactor Site Criteria,” requires that SSCs necessary to assure the capability of the plant to mitigate the consequences of accidents, which could result in exposures comparable to the guideline exposures provided in 10 CFR Part 100, be designed to remain functional during and after a safe shutdown earthquake. In accordance with the 10 CFR Part 100 requirements, please identify SSCs which may be affected by the implementation of the proposed AST and address the following:

a) Indicate whether any non-safety-related SSCs are being credited in the proposed AST license amendment.

b) For any nonsafety-related SSCs credited in the AST, confirm that the SSCs have been seismically qualified in accordance with the plant licensing basis.

c) Indicate whether the SSCs are new or existing.

d) Describe the location of the SSCs and the seismic qualification method employed to demonstrate the seismic ruggedness of these SSCs, such as the plant licensing basis or an NRC-endorsed industry standard.

e) Summarize the results of the seismic qualification of the equipment, indicating whether any modifications or re-design will be necessary in support of the AST.

Response:

The proposed license amendment request included the adoption of Alternative Source Term (AST) radiological analysis methodology consistent with 10 CFR 50.67, “Accident source term.” The design basis radiological consequence analyses performed for AST did not credit any non-safety related structures, systems, and components (SSCs). This is consistent with the current licensing basis analysis of record that does not credit any non-safety related SSCs. **The design basis radiological consequences analyses performed for AST do not require the installation of any new SSCs (the damper modification being implemented will only change the damper power supply; it will not add new components). New cable trays or conduit may need to be installed to support the one modification being implemented, depending on cable tray and conduit availability. If the modification requires installation of a new cable tray or conduit for the power cables, the associated seismic evaluations will be performed in accordance with Wolf Creek methods used for existing cable trays and conduit.**

ESGB-RAI-1

Describe the analysis methodology used to determine the pH in the sump water during the period of 30 days post-LOCA. Include detailed calculations of time dependent pH values in the sump during a 30 day period post-LOCA to demonstrate that the pH remains basic throughout this time period.

Response:

As discussed in Section 4.4.2 of Enclosure VI to Reference 1, the containment sump pH calculation did not include consideration of acid generation (nitric acid produced by the irradiation of water and air or hydrochloric acid produced by the radiolysis of chlorine bearing materials) as they were considered secondary effects. In Section 4.4.2 of Enclosure VI to Reference 1, WCNOC indicated that it was expected that the effect of acid generation on sump pH would decrease the pH value less than 0.1 pH based on a comparison of Byron Station information and the conclusion reached by the NRC in the safety evaluation for Amendment No. 147 for the Byron Station and Amendment No. 140 for the Braidwood Station (Reference 23).

WCNOC letter WO 14-0032 (Reference 22) provided the below details and results of the calculation.

Methodology

In order to calculate the minimum sump pH, the maximum amount of boric acid from the various sources of borated water that enter the containment sump and the strong acids that are generated from the effects of radiolysis post-LOCA combined with a minimum amount of caustic from the spray additive tank (SAT) will yield a minimum pH value. The concentrations of these substances are used to compute the value of the sump pH as a function of time using verified titration curve data for aqueous solutions of boric acid and sodium hydroxide.

The current licensing basis (CLB) for the WCGS sump pH calculation (EN-03-W, Rev. 2, “18 Months Fuel Cycle, Cycle 4 Specific Boron – pH Calculations for TSA 20038-001, Rev. 3”) forms the basis of existing Updated Safety Analysis Report (USAR) Figure 6.5-5. Two scenarios are analyzed: both containment spray trains operating with one NaOH eductor in service, and both containment spray trains operating with both NaOH eductors in service.

No computer codes were used in the CLB calculation. To determine the pH in this calculation, NaOH and H_3BO_3 molarities were first calculated and then used with Oak Ridge National Laboratory (ORNL) titration curve data to determine the pH of the spray and sump solutions during the injection and recirculation phases of Emergency Core Cooling System (ECCS) operation. The CLB calculation did not consider the effects of the generation of strong acids inside containment post-LOCA.

Westinghouse has subsequently performed an independent calculation (CN-SEE-I-13-10, Rev. 1, “Wolf Creek Sump pH,”) to update the conclusions of the CLB calculation for the transition to the Westinghouse Core Design and Safety Analysis methodologies. The Westinghouse calculation used pH data for boric acid/sodium hydroxide solutions from verified titration curve data for aqueous solutions of boric acid and sodium hydroxide. A revision of this calculation was performed in response to this RAI question (ESGB-RAI-1) and the effect of strong acid generation on the post-LOCA sump pH was considered. In accordance with the guidance of NUREG/CR-5950, “Iodine Evolution and pH Control,” the mass of nitric acid that is generated by the radiolysis of air and water inside containment and the mass of hydrochloric acid that is generated by the radiolysis of Hypalon® electrical cable insulation inside containment were calculated. The amount of hydriodic acid that is generated from iodine released inside containment was judged to be negligible and was not considered in the sump pH calculation.

The masses (moles) of the strong acids that are generated inside containment post-LOCA were assumed to instantaneously be neutralized by a molar equivalent of NaOH, which reduced the net mass of NaOH that was available to neutralize the boric acid injected from the Reactor Coolant System (RCS), accumulators and refueling water storage tank (RWST). The net effect of the neutralization of all acid species in the sump by the NaOH from the SAT determined the equilibrium sump pH.

The CLB has been updated since the time the RAIs were sent and thus the current USAR figure is aligned with the AST analysis.

Assumptions and Inputs

Among the various assumptions that were made in the sump pH calculation is that of effective mixing of the water inventory in the sump. This is a valid assumption because of the long term operation of the Containment Spray System and the uniform dispersion of spray over the containment cross section. With respect to the generation of strong acids inside containment, the gaseous hydrochloric acid produced by electric cable insulation radiolysis was conservatively assumed to be instantly dissolved in the sump water, and the hydrochloric acid generated from the radiolysis of cable insulation and nitric acid generated by the irradiation of containment sump water and air was assumed to fully dissociate in the sump water. At the completion of the injection and mixing of all chemical species (acids and sodium hydroxide) inside containment post-LOCA, it was assumed that there would be no further change in the

sump liquid inventory and, consequently, that there would be no further change in the calculated sump pH endpoint over the 30 day post-LOCA period.

The minimum long-term sump pH in the revised Westinghouse calculation uses the following conservative bases:

7. Maximum RWST and safety injection (SI) accumulator Technical Specification 3.5.1, “Accumulators,” and 3.5.4, “Refueling Water Storage Tank (RWST),” boron concentration of 2500 ppm.

8. A conservatively high RCS boron concentration of 1900 ppm in the CLB, and 1980 ppm in the Westinghouse calculation. Given that the RCS is only approximately 15% of the sump fluid, this difference in boron concentration is a small effect.

9. The SAT is assumed to contain the Technical Specification 3.6.7, “Spray Additive System,” minimum 28% by weight NaOH solution.

10. The mass of Hypalon® electrical cable insulation inside containment is 50,000 lb_m.

11. The 30-day integrated containment doses inside containment (beta and gamma sources) to which the Hypalon® insulation is exposed post-LOCA were derived from the plant-specific report for the Environmental Qualification of Safety-Related Electrical Equipment.

12. The maximum sump liquid mass is approximately $4.1x10^6$ lb_m, which corresponds to a liquid volume of $1.87x10^6$ liters.

While the Technical Specification 3.6.7 SAT minimum contained volume is 4340 gallons, the CLB analysis assumed a conservative minimum delivered volume of only 2960 gallons. The Westinghouse calculation used an adjusted delivered volume of 2752 gallons, which accounted for the neutralization of the strong acid generated inside containment post-LOCA.

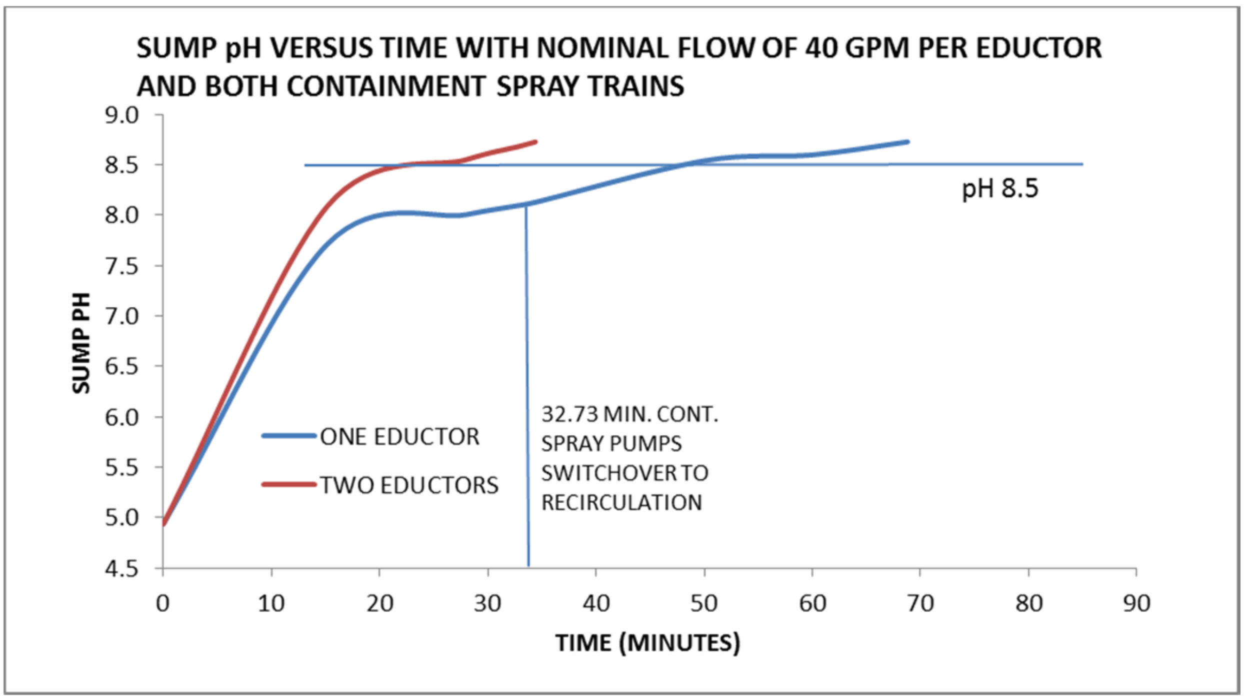
Results and Conclusions

The Westinghouse calculation determined a long-term sump pH of 8.7, while the CLB calculation determined a minimum value of 8.6. The slightly higher pH in the Westinghouse calculation is attributed to the use of a different boron/NaOH/pH correlation. In either case, the results show that the sump pH remains above the CLB value of 8.5 and well above the NRC-required value of 7.0 after the minimum amount of NaOH is injected into the sump.

In the CLB calculation, the limiting system alignment of both containment spray trains operating with one NaOH eductor in service results in a sump pH of 7 in approximately 15 minutes, with the final long-term pH of 8.6 in approximately 80 minutes. The less limiting case of both Containment Spray trains operating with both NaOH supplies in service results in a sump pH of 7 in approximately 11 minutes, with the final long-term pH of 8.6 in approximately 45 minutes.

For the Westinghouse calculation, the limiting system alignment of both containment spray trains operating with one NaOH eductor in service results in a sump pH of 7 in approximately 11 minutes, with the final long-term pH of 8.7 in approximately 70 minutes. The less limiting case of both containment spray trains operating with both NaOH supplies in service results in a sump pH of 7 in approximately 9 minutes, with the final long-term pH of 8.7 in approximately 35 minutes.

The figure below from the Wolf Creek USAR shows that the acceptance criteria of sump pH > 8.5 (minimum) is met for one or two containment spray eductors in service, considering also the generation of strong acids inside containment post-LOCA, consistent with the guidance of NUREG/CR-5950.



WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.5-5

CONTAINMENT SUMP PH WITH NOMINAL EDUCTOR FLOW FOR ONE AND TWO EDUCTOR OPERATION

Reference 25 RAIs

<u>EEEB-RAI-1</u>

Please confirm whether any loads are being added to the WCGS emergency diesel generators (EDGs). If so, describe their impact on the capability and capacity of the EDGs. Also, describe changes, if any, being made to the EDG loading sequence to support this license amendment request (LAR).

Response:

The proposed license amendment request included the adoption of Alternative Source Term (AST) radiological analysis methodology consistent with 10 CFR 50.67, “Accident source term.” The design basis radiological consequence analyses performed for AST did not result in the addition of loads to the onsite standby power source (train A and train B diesel generators (DGs)) and did not result in changes to the DG loading sequence.

<u>EEEB-RAI-2</u>

Please confirm whether any components are being added to the Environmental Qualification (EQ) equipment list to comply with Title 10 of the Code of Federal Regulations, Section 50.49 (10 CFR 50.49) due to this LAR. If components are being added, describe the equipment qualification for the environmental conditions to which the components are expected to be exposed.

Response:

The design basis radiological consequences analyses performed for AST do not require the installation of any new SSCs. The one modification being implemented to support full implementation of AST will not add any components to the EQ list (the damper modification being implemented will only change the damper power supply; it will not add new components). As such, no components are added to the Environmental Qualification equipment list to comply with 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants.”

Reference 2 RAIs

<u>ARCB-RAI-1</u>

Please justify all changes from the current licensing basis (see Issue 1 of NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms," dated March 7, 2006 (ADAMS Accession No. ML053460347), for more detail). No justification is needed for changes that are consistent with Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792), or are provided in the submittal dated August 13, 2013 (ADAMS Accession No. ML 13247A076), unless requested by these RAIs.

Response:

NRC Regulatory Issue Summary 2006-004, "Experience with Implementation of Alternative Source Terms," indicates that 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. Section 1 of Enclosure VI of Reference 1 identified the changes to the WCGS licensing basis as:

- The Control Room Habitability Envelope (CRHE) unfiltered inleakage is revised from 20 scfm to 50 scfm.

- The Control Building unfiltered inleakage is revised from 300 scfm to 400 scfm.

- Revise the USAR Chapter 15 dose analyses in accordance with the guidance in Regulatory Guide 1.183.

- Revise the Technical Specification (TS) to address the update of the accident source term and associated design basis accidents utilizing the guidance provided in Regulatory Guide 1.183 and the associated control room dose limits of General Design Criterion (GDC) 19, and offsite dose limits of 10 CFR 50.67.

In addition, a detailed comparison of the analyses parameters for the AST versus the current licensing basis is included in Enclosure IV of this LAR, Section 4.3, Tables 4.3-5 through 4.3-16. These tables have been included below to facilitate the review. For the resubmittal of the LAR, the justification for the proposed changes to the TSs and the control room and control building unfiltered inleakages is justified in Section 2 of Enclosure IV.

Doses Analyses Parameters Summary Tables (excerpt from Enclosure IV)

Table 4.3-5 Control Room and Control Building Parameters

	AST	CLB	Reason for Change
Control room volume (ft^3)	100,000	100,000	No change
Control building volume (ft^3)	239,000	239,000	No change
Normal ventilation flow rates (cfm)			
Unfiltered makeup flow rate from environment to control building	13,050	13,050	No change
Unfiltered makeup flow rate from environment to control room	1950	1950	No change
Unfiltered inleakage to control room	50	10	Increased to allow for additional surveillance testing margin
Emergency mode of operation flow rates prior to operator action (cfm)			
Filtered makeup flow rate from environment to control building	1350	1350	No change
Filtered makeup flow rate from control building to control room	550	550	No change
Unfiltered makeup flow rate from environment to control building	400	300	Increased to allow for additional surveillance testing margin
Unfiltered makeup flow rate from control building to control room	550	550	No change
Unfiltered inleakage to control room	50	20	Increased to allow for additional surveillance testing margin
Filtered control room recirculation flow	1250	1250	No change
Emergency mode of operation flow rates following operator action (cfm)			
Filtered makeup flow rate from environment to control building	675	675	No change
Filtered makeup flow rate from control building to control room	550	550	No change
Unfiltered makeup flow rate from environment to control building	400	300	Increased to allow for additional surveillance testing margin

Table 4.3-5 Control Room and Control Building Parameters (cont.)

	AST	CLB	Reason for Change
Unfiltered makeup flow rate from control building to control room	0	0	No change
Unfiltered inleakage to control room	50	20	Increased to allow for additional surveillance testing margin
Filtered control room recirculation flow	1250	1250	No change
Operator action time to terminate failed train of filtered makeup flow from start of event (minutes)	90	90	No change
Filter efficiencies (%)			
Elemental iodine	95	95	No change
Organic iodine	95	95	No change
Particulates	95	95	No change
Isolation setpoint for control room air supply radiation monitors (GKRE0004 and GKRE0005) (μCi/cc Xe-133)	2.12E-03	N/A	Change due to modeling of CR for all accidents (not AST specific)
Delay to switch to emergency mode of operation following receipt of isolation signal (seconds)	60	N/A	Change due to modeling of CR for all accidents (not AST specific)
Control room breathing rate for duration of the event (m^3/sec)	3.5E-04	3.47E-04	Regulatory Guide update (not AST specific)
Control room occupancy factors			
0 – 24 hours	1.0	1.0	No change
1 – 4 days	0.6	0.6	No change
4 – 30 days	0.4	0.4	No change

Table 4.3-6 Assumptions Used for Main Steamline Break Analysis

	AST	CLB	Reason for Change
RCS activity	See Table 4.3-1a	See Table 4.3-1b	Updated calculations
Initial secondary system activity	See Table 4.3-1a	See Table 4.3-1b	Updated calculations
Pre-accident iodine spike factor	60	60	No change
Accident-initiated iodine spike appearance rate calculations			
Letdown flow, maximum (gpm)	132	75	Modeling update (not AST specific)
Letdown flow decontamination (%)	100	N/A	Modeling update (not AST specific)
RCS leakage (gpm)	11	1	Modeling update (not AST specific)
Spike factor	500	500	No change
Duration of accident-initiated iodine spike (hr)	8	8	No change
RCS mass, maximum (lbm)	8.42E+05	4.94E+05	Modeling update (not AST specific)
Equilibrium appearance rates (Ci/min)			
I-130	9.87E-03	N/A	Modeling update (not AST specific)
I-131	4.39E-01	N/A	Modeling update (not AST specific)
I-132	1.98E+00	N/A	Modeling update (not AST specific)
I-133	8.93E-01	N/A	Modeling update (not AST specific)
I-134	9.65E-01	N/A	Modeling update (not AST specific)
I-135	8.17E-01	N/A	Modeling update (not AST specific)

Table 4.3-6 Assumptions Used for Main Steamline Break Analysis (cont.)			
	AST	**CLB**	**Reason for Change**
Iodine chemical form of releases (%)			
Elemental	97	N/A	Regulatory Guide Update (not AST specific)
Organic	3	N/A	Regulatory Guide Update (not AST specific)
Particulate	0	N/A	Regulatory Guide Update (not AST specific)
Approximate timing of events			
Safety injection (SI) signal (sec)	30	N/A	Modeling update (not AST specific)
Control room isolation (including delay) (sec)	90	N/A	Modeling update (not AST specific)
Faulted SG releases all initial activity (min)	2	N/A	Modeling update (not AST specific)
RHR cooling takes over (releases from intact SGs terminated) (hr)	12	N/A	Modeling update (not AST specific)
RCS cooled below 212°F (releases from faulted SG terminated) (hr)	34	N/A	Modeling update (not AST specific)
Mass transfer data			
Initial faulted SG release (in first 2 minutes) (lbm)	165,000	164,500	Updated calculations
Total primary-to-secondary leakage			
Leakage through faulted SG to atmosphere (gpm)	1	1	No change
Leakage into intact SGs (gpd, total)	450	N/A	Modeling update (not AST specific)
Steam Released from Intact SGs to Atmosphere			
0 to 2 hours (lbm)	419,340	404,452	Updated calculations
2 to 12 hours (lbm)	1,310,269	945,973	Updated calculations

Table 4.3-6 Assumptions Used for Main Steamline Break Analysis (cont.)

	AST	CLB	Reason for Change
RCS Mass, Minimum (lbm)	3.99E+05	4.94E+05 CLB does not use maximum and minimum	Input change (minimum vs nominal)
Faulted SG Mass, Maximum (lbm)	1.65E+05	164,500	Updated calculations
Intact SGs Mass, Minimum (lbm, Total)	2.47E+05	286,500	Updated calculations
SG iodine water/steam partition coefficient	100	100	No change
Moisture carryover (%)	0.25	0.25	No change
Control room atmospheric dispersion factors (sec/m^3)			
Intact SGs			
0 – 0.025 hours	2.55E-02	N/A	Not modeled in current AORs
0.025 – 2 hours	1.04E-03	N/A	Not modeled in current AORs
2 – 8 hours	7.46E-04	N/A	Not modeled in current AORs
8 – 24 hours	3.03E-04	N/A	Not modeled in current AORs
24 – 96 hours	1.90E-04	N/A	Not modeled in current AORs
96 – 720 hours	1.39E-04	N/A	Not modeled in current AORs
Faulted SG			
0 – 0.025 hours	2.11E-03	N/A	Not modeled in current AORs
0.025 – 2 hours	6.12E-04	N/A	Not modeled in current AORs
2 – 8 hours	4.38E-04	N/A	Not modeled in current AORs
8 – 24 hours	1.79E-04	N/A	Not modeled in current AORs
24 – 96 hours	1.14E-04	N/A	Not modeled in current AORs

Table 4.3-6 Assumptions Used for Main Steamline Break Analysis (cont.)

	AST	CLB	Reason for Change
96 – 720 hours	8.94E-05	N/A	Not modeled in current AORs
TSC atmospheric dispersion factors (sec/m^3)			
Intact SGs			
0 – 2 hours	4.83E-04	N/A	Not modeled in current AORs
2 – 8 hours	2.58E-04	N/A	Not modeled in current AORs
8 – 24 hours	9.63E-05	N/A	Not modeled in current AORs
24 – 96 hours	6.45E-05	N/A	Not modeled in current AORs
96 – 720 hours	4.89E-05	N/A	Not modeled in current AORs
Faulted SGs			
0 – 2 hours	2.80E-04	N/A	Not modeled in current AORs
2 – 8 hours	1.80E-04	N/A	Not modeled in current AORs
8 – 24 hours	6.44E-05	N/A	Not modeled in current AORs
24 – 96 hours	4.42E-05	N/A	Not modeled in current AORs
96 – 720 hours	3.22E-05	N/A	Not modeled in current AORs

Table 4.3-7 Assumptions Used for Loss of Non-Emergency AC Power Analysis			
	AST	**CLB**	**Reason for Change**
RCS activity	See Table 4.3-1a	See Table 4.3-1b	Updated calculations
Initial secondary system activity	See Table 4.3-1a	See Table 4.3-1b	Updated calculations
Accident-initiated iodine spike appearance rate calculations			
Letdown flow, maximum (gpm)	132	75	Modeling update (not AST specific)
Letdown flow decontamination (%)	100	N/A	Modeling update (not AST specific)
RCS leakage (gpm)	11	1	Modeling update (not AST specific)
Spike factor	500	N/A	Modeling update (not AST specific)
Duration of accident-initiated iodine spike (hr)	8	N/A	Modeling update (not AST specific)
RCS mass, maximum (lbm)	8.42E+05	4.94E+05 CLB does not use max and min	Input change (maximum vs nominal)
Equilibrium appearance rates (Ci/min)			
I-130	9.87E-03	N/A	Modeling update (not AST specific)
I-131	4.39E-01	N/A	Modeling update (not AST specific)
I-132	1.98E+00	N/A	Modeling update (not AST specific)
I-133	8.93E-01	N/A	Modeling update (not AST specific)
I-134	9.65E-01	N/A	Modeling update (not AST specific)
I-135	8.17E-01	N/A	Modeling update (not AST specific)

Table 4.3-7 Assumptions Used for Loss of Non-Emergency AC Power Analysis (cont.)

	AST	CLB	Reason for Change
Iodine chemical form of releases (%)			
Elemental	97	N/A	Regulatory Guide Update (not AST specific)
Organic	3	N/A	Regulatory Guide Update (not AST specific)
Particulate	0	N/A	Regulatory Guide Update (not AST specific)
Time RHR cooling matched decay heat (SG releases terminated) (hr)	12	8	Updated calculations
Mass transfer data			
Total primary-to-secondary leakage (gpm)	1	1	No change
Steam released from SGs to atmosphere			
0 to 2 hours (lbm)	419,846	549,000	Updated calculations
2 to 12 hours (lbm)	1,352,918	1,030,000	Updated calculations
RCS mass, minimum (lbm)	3.99E+05	4.94E+05 CLB does not use max and min	Input change (minimum vs nominal)
Plant total SG mass, minimum (lbm)			
Until 2 hours (lbm)	3.30E+05	382,000	Updated calculations
After 2 hours (lbm)	4.85E+05	382,000	Modeling update (not AST specific)
SG iodine water/steam partition coefficient	100	100	No change
Moisture carryover (%)	0.25	0.25	No change
Control room isolation	None	N/A	No change

Table 4.3-7 Assumptions Used for Loss of Non-Emergency AC Power Analysis (cont.)

	AST	CLB	Reason for Change
Control room atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	2.55E-2	N/A	Not modeled in current AORs
2 – 8 hours	1.84E-2	N/A	Not modeled in current AORs
8 – 24 hours	7.46E-3	N/A	Not modeled in current AORs
24 – 96 hours	4.72E-3	N/A	Not modeled in current AORs
96 – 720 hours	3.43E-3	N/A	Not modeled in current AORs
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	4.83E-04	N/A	Not modeled in current AORs
2 – 8 hours	2.58E-04	N/A	Not modeled in current AORs
8 – 24 hours	9.63E-05	N/A	Not modeled in current AORs
24 – 96 hours	6.45E-05	N/A	Not modeled in current AORs
96 – 720 hours	4.89E-05	N/A	Not modeled in current AORs

Table 4.3-8 Assumptions Used for Locked Rotor Analysis

	AST	CLB	Reason for Change
Core activity	See Table 4.3-1a	See Table 4.3-1b	Updated calculations
Failed fuel (% of Core)	5	5	No change
Melted fuel (% of Core)	0	0	No change
Peaking factor	1.65	1.65	No change
Gap fractions			
I-131	0.08	0.12	Modeling change for AST
Kr-85	0.10	0.30	Modeling change for AST
Other iodines and noble gases	0.05	0.10	Modeling change for AST
Iodine chemical form of releases (%)			
Elemental	97	N/A	Regulatory Guide Update (not AST specific)
Organic	3	N/A	Regulatory Guide Update (not AST specific)
Particulate	0	N/A	Regulatory Guide Update (not AST specific)
Time RHR cooling matched decay heat (SG releases terminated) (hr)	12	8	Updated calculations
Mass transfer data			
Total primary-to-secondary leakage (gpm)	1	1	No change
Steam released from SGs to atmosphere			
0 to 2 hours (lbm)	419,846	5.49E+05	Updated calculations
2 to 12 hours (lbm)	1,352,918	1.03E+06	Updated calculations
RCS mass, minimum (lbm)	3.99E+05	4.94E+05	Input change (minimum vs nominal)
Plant total SG mass, minimum (lbm)			
Until 2 hours (lbm)	3.30E+05	3.82E+05	Updated calculations
After 2 hours (lbm)	4.85E+05	3.82E+05	Updated calculations

Table 4.3-8 Assumptions Used for Locked Rotor Analysis (cont.)

	AST	CLB	Reason for Change
SG iodine water/steam partition coefficient	100	100	No change
Moisture carryover (%)	0.25	0.25	No change
Time of Control room isolation (including delays) (sec)	120	N/A	No change
Control room atmospheric dispersion factors (sec/m^3)			
HVAC Flows Except for Control Room Unfiltered Inleakage			
0 – 0.0333 hours	2.55E-02	N/A	Not modeled in current AORs
0.0333 – 2 hours	1.04E-03	N/A	Not modeled in current AORs
2 – 8 hours	7.46E-04	N/A	Not modeled in current AORs
8 – 24 hours	3.03E-04	N/A	Not modeled in current AORs
24 – 96 hours	1.90E-04	N/A	Not modeled in current AORs
96 – 720 hours	1.39E-04	N/A	Not modeled in current AORs
Control Room Unfiltered Inleakage			
0 – 2 hours	2.55E-02	N/A	Not modeled in current AORs
2 – 8 hours	1.84E-02	N/A	Not modeled in current AORs
8 – 24 hours	7.46E-03	N/A	Not modeled in current AORs
24 – 96 hours	4.72E-03	N/A	Not modeled in current AORs
96 – 720 hours	3.43E-03	N/A	Not modeled in current AORs

Table 4.3-8 Assumptions Used for Locked Rotor Analysis (cont.)

	AST	CLB	Reason for Change
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	4.83E-04	N/A	Not modeled in current AORs
2 – 8 hours	2.58E-04	N/A	Not modeled in current AORs
8 – 24 hours	9.63E-05	N/A	Not modeled in current AORs
24 – 96 hours	6.45E-05	N/A	Not modeled in current AORs
96 – 720 hours	4.89E-05	N/A	Not modeled in current AORs

Table 4.3-9 Assumptions Used for Rod Ejection Analysis

	AST	CLB	Reason for Change
Core activity	See Table 4.3-1a	See Table 4.3-1b	Updated calculations
Failed fuel (% of core)	10	10	No change
Melted fuel (% of core)	0.25	0.25	No change
Peaking factor	1.65	1.65	No change
Gap fractions			
Iodines and noble gases	0.10	0.10	No change
Alkali metals	0.12	N/A	Modeling change for AST
Containment Leakage			
Activity released to containment from failed fuel (%)			
Iodines and noble gases	10	10	No change
Alkali metals	12	N/A	Modeling change for AST
Activity released to containment from melted fuel (%)			
Iodines and alkali metals	50	50	No change
Noble gas	100	100	No change
Iodine chemical form of releases (%)			
Elemental	4.85	91	Modeling change for AST
Particulate	95	5	Modeling change for AST
Organic	0.15	4	Modeling change for AST
Containment leak rates (weight %/day)			
0 – 24 hours	0.2	0.2	No change
1 – 30 days	0.1	0.1	No change
Containment volume (ft^3)	2.5E+06	2.5E+06	No change
Removal of airborne activity in containment (other than leakage or decay)	None	None	No change

Table 4.3-9 Assumptions Used for Rod Ejection Analysis (cont.)			
	AST	**CLB**	**Reason for Change**
SI signal (sec)	150	N/A	Modeling update (not AST specific)
Time of control room isolation (including delays) (sec)	210	N/A	Modeling update (not AST specific)
Control room atmospheric dispersion factors (sec/m^3)			
0 – 0.0583 hours	2.11E-03	N/A	Not modeled in current AORs
0.0583 – 2 hours	6.12E-04	N/A	Not modeled in current AORs
2 – 8 hours	4.38E-04	N/A	Not modeled in current AORs
8 – 24 hours	1.81E-04	N/A	Not modeled in current AORs
24 – 96 hours	1.29E-04	N/A	Not modeled in current AORs
96 – 720 hours	9.65E-05	N/A	Not modeled in current AORs
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	3.91E-04	N/A	Not modeled in current AORs
2 – 8 hours	2.66E-04	N/A	Not modeled in current AORs
8 – 24 hours	9.62E-05	N/A	Not modeled in current AORs
24 – 96 hours	7.05E-05	N/A	Not modeled in current AORs
96 – 720 hours	5.52E-05	N/A	Not modeled in current AORs
Primary-to-Secondary Leakage			
Activity released to RCS from failed fuel (%)			
Iodines and noble gases	10	10	No change
Alkali metals	12	N/A	Modeling change for AST

Table 4.3-9 Assumptions Used for Rod Ejection Analysis (cont.)

	AST	CLB	Reason for Change
Activity released to RCS from melted fuel (%)			
Iodines and alkali metals	50	50	No change
Noble gas	100	100	No change
Iodine chemical form of releases (%)			
Elemental	97	N/A	Regulatory Guide Update (not AST specific)
Organic	3	N/A	Regulatory Guide Update (not AST specific)
Particulate	0	N/A	Regulatory Guide Update (not AST specific)
Time RHR cooling matched decay heat (SG releases terminated) (hr)	12	8	Updated calculations
Mass transfer data			
Total primary-to-secondary leakage (gpm)	1	1	No change
Steam released from SGs to atmosphere			
0 to 2 hours (lbm)	419,846	48,600 (140 sec)	Updated calculations
2 to 12 hours (lbm)	1,352,918	N/A	Updated calculations
RCS mass, minimum (lbm)	3.99E+05	4.94E+05	Input change (minimum vs nominal)
Plant total SG mass, minimum (lbm)			
Until 2 hours (lbm)	3.30E+05	4.16E+05	Updated calculations
After 2 hours (lbm)	4.85E+05	4.16E+05	Updated calculations

Table 4.3-9 Assumptions Used for Rod Ejection Analysis (cont.)

	AST	CLB	Reason for Change
SG iodine water/steam partition coefficient	100	100	No change
Moisture carryover (%)	0.25	0.25	No change
Time of Control room isolation (including delays) (sec)	120	N/A	No change
Control room atmospheric dispersion factors (sec/m^3)			
HVAC Flows Except for Control Room Unfiltered Inleakage			
0 – 0.0333 hours	2.55E-02	N/A	Not modeled in current AORs
0.0333 – 2 hours	1.04E-03	N/A	Not modeled in current AORs
2 – 8 hours	7.46E-04	N/A	Not modeled in current AORs
8 – 24 hours	3.03E-04	N/A	Not modeled in current AORs
24 – 96 hours	1.90E-04	N/A	Not modeled in current AORs
96 – 720 hours	1.39E-04	N/A	Not modeled in current AORs
Control Room Unfiltered Inleakage			
0 – 2 hours	2.55E-02	N/A	Not modeled in current AORs
2 – 8 hours	1.84E-02	N/A	Not modeled in current AORs
8 – 24 hours	7.46E-03	N/A	Not modeled in current AORs
24 – 96 hours	4.72E-03	N/A	Not modeled in current AORs
96 – 720 hours	3.43E-03	N/A	Not modeled in current AORs
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	4.83E-04	N/A	Not modeled in current AORs

Table 4.3-9 Assumptions Used for Rod Ejection Analysis (cont.)

	AST	CLB	Reason for Change
2 – 8 hours	2.58E-04	N/A	Not modeled in current AORs
8 – 24 hours	9.63E-05	N/A	Not modeled in current AORs
24 – 96 hours	6.45E-05	N/A	Not modeled in current AORs
96 – 720 hours	4.89E-05	N/A	Not modeled in current AORs

Table 4.3-10 Assumptions Used for Letdown Line Break Analysis

	AST	CLB	Reason for Change
RCS activity	See Table 4.3-1a	See Table 4.3-1b	Updated calculations
Accident-initiated iodine spike appearance rate calculations			
Letdown flow, maximum (gpm)	132	195	Modeling update (not AST specific)
Letdown flow decontamination (%)	100	N/A	Modeling update (not AST specific)
RCS leakage (gpm)	11	N/A	Modeling update (not AST specific)
Spike factor	500	N/A	Modeling update (not AST specific)
Duration of accident-initiated iodine spike (hr)	8	N/A	Modeling update (not AST specific)
Reactor coolant mass, maximum (lbm)	8.42E+05	4.94E+05	Input change (maximum vs nominal)
Equilibrium appearance rates (Ci/min)			
I-130	9.87E-03	N/A	Modeling update (not AST specific)
I-131	4.39E-01	N/A	Modeling update (not AST specific)
I-132	1.98E+00	N/A	Modeling update (not AST specific)
I-133	8.93E-01	N/A	Modeling update (not AST specific)
I-134	9.65E-01	N/A	Modeling update (not AST specific)
I-135	8.17E-01	N/A	Modeling update (not AST specific)

Table 4.3-10 Assumptions Used for Letdown Line Break Analysis (cont.)

	AST	CLB	Reason for Change
Iodine chemical form of releases (%)			
Elemental	97	N/A	Regulatory Guide Update (not AST specific)
Organic	3	N/A	Regulatory Guide Update (not AST specific)
Particulate	0	N/A	Regulatory Guide Update (not AST specific)
Reactor coolant mass, minimum (lbm)	3.99E+05	N/A	Modeling update (not AST specific)
Flow rate out of broken line (gpm)	141	141	No change
Iodine and alkali metal airborne fraction	0.18	N/A	Modeling update (not AST specific)
Maximum RCS letdown pressure (psig)	600	2200	Modeling update (not AST specific)
Maximum RCS letdown temperature (°F)	380	286	Modeling update (not AST specific)
Time to isolate break flow (terminating releases) (min)	30.167	30.167	No change
Control room isolation	None	N/A	No change
Control room atmospheric dispersion factor (sec/m^3)		N/A	
0 – 2 hours	2.11E-03	N/A	Not modeled in current AORs
TSC atmospheric dispersion factor (sec/m^3)			
0 – 2 hours	2.80E-04	N/A	Not modeled in current AORs

Table 4.3-11 Assumptions Used for SGTR Dose Analysis

	AST	CLB	Reason for Change
RCS activity	See Table 4.3-1a	See Table 4.3-1c	Updated calculations
Initial secondary system activity	See Table 4.3-1a	See Table 4.3-1c	Updated calculations
Pre-accident iodine spike factor	60	60	No change
Accident-initiated iodine spike appearance rate calculations			
Letdown flow, maximum (gpm)	132	120	Modeling update (not AST specific)
Letdown flow decontamination (%)	100	N/A	Modeling update (not AST specific)
RCS leakage (gpm)	11	1	Modeling update (not AST specific)
Spike factor	335	335	No change
Reactor coolant mass, maximum (lbm)	8.42E+05	5.05E+5	Input change (maximum vs nominal)
Duration of accident-initiated iodine spike (hr)	8	8	No change
Equilibrium appearance rates (Ci/min)			
I-130	9.87E-03	N/A	Modeling update (not AST specific)
I-131	4.39E-01	3.54E-1	Modeling update (not AST specific)
I-132	1.98E+00	1.36E+00	Modeling update (not AST specific)
I-133	8.93E-01	7.72E+00	Modeling update (not AST specific)
I-134	9.65E-01	7.02E-01	Modeling update (not AST specific)
I-135	8.17E-01	6.63E-01	Modeling update (not AST specific)

Table 4.3-11 Assumptions Used for SGTR Dose Analysis (cont.)

	AST	CLB	Reason for Change
Iodine chemical form of releases (%)			
Elemental	97	N/A	Regulatory Guide Update (not AST specific)
Organic	3	N/A	Regulatory Guide Update (not AST specific)
Particulate	0	N/A	Regulatory Guide Update (not AST specific)
Approximate timing of events (sec)	See Table 2.7.3-2 in Enclosure I of this LAR (For dose input, exclude the 100 seconds of steady-state operation.)	N/A	Modeling update (not AST specific)
Time of control room isolation (including delay) (sec)	120	N/A	Modeling update (not AST specific)
Time to complete control room isolation with SI signal from event initiation (sec)	600	N/A	Modeling update (not AST specific)

Table 4.3-11 Assumptions Used for SGTR Dose Analysis (cont.)

	AST	CLB	Reason for Change
Transient mass transfer data			
Non-flashed break flow (lbm)			
0 – 52 seconds	2227.5	N/A	Updated calculations
52 – 1102 seconds	43,129.9	N/A	Updated calculations
1102 – 2902 seconds	88,387.2	N/A	Updated calculations
2902 – 3502 seconds	32,991.2	N/A	Updated calculations
3502 – 3846 seconds	18,224.8	N/A	Updated calculations
3846 – 5155 seconds	61,523.0	N/A	Updated calculations
5155 – 7527 seconds	41,166.4	N/A	Updated calculations
Flashed break flow (lbm)			
0 – 52 seconds	438.9	N/A	Updated calculations
52 – 1102 seconds	2,901.8	N/A	Updated calculations
1102 – 2902 seconds	13,432.1	N/A	Update calculations
2902 – 3502 seconds	2,635.6	N/A	Update calculations
3502 – 3846 seconds	606.1	N/A	Update calculations

Table 4.3-11 Assumptions Used for SGTR Dose Analysis (cont.)

	AST	CLB	Reason for Change
Steam released from ruptured SG (lbm)			
0 – 52 seconds	188,100	N/A	Update calculations
52 – 1102 seconds	27,469.2	N/A	Update calculations
1102 – 2902 seconds	149,850.8	N/A	Update calculations
2902 – 7527 seconds	0	N/A	Update calculations
7527 – 43,200 seconds	2530	N/A	Update calculations
Steam released from intact SGs (lbm)			
0 – 52 seconds	562,650	N/A	Update calculations
52 – 1102 seconds	69,877.5	N/A	Update calculations
1102 – 3502 seconds	0	N/A	Update calculations
3502 – 3846 seconds	94,307.4	N/A	Update calculations
3846 – 5155 seconds	130,799.9	N/A	Update calculations
5155 – 7527 seconds	98,156.3	N/A	Update calculations
7527 – 43,200 seconds	1,645,930	N/A	Update calculations
Reactor coolant mass, minimum (lbm)	3.99E+05	5.05E+05 CLB does not use max and min	Input change (minimum vs nominal)
Ruptured SG mass, minimum (lbm)	7.00E+04	N/A	Updated calculations
Intact SGs mass, minimum (lbm, total)	1.95E+05	N/A	Updated calculations
Condenser iodine and alkali metal removal factor	100	N/A	Modeling update (not AST specific)
SG iodine water/steam partition coefficient	100	N/A	Regulatory Guide Update (not AST specific)
Moisture carryover (%)	0.25	0.25	No change

Table 4.3-11 Assumptions Used for SGTR Dose Analysis (cont.)

	AST	CLB	Reason for Change
Control room atmospheric dispersion factors (sec/m^3)			
0 – 0.0333 hours	2.55E-02	N/A	Not modeled in current AORs
0.0333 – 2 hours$^{(1)}$	1.04E-03	N/A	Not modeled in current AORs
2 – 8 hours	7.46E-04	N/A	Not modeled in current AORs
8 – 24 hours	3.03E-04	N/A	Not modeled in current AORs
24 – 96 hours	1.90E-04	N/A	Not modeled in current AORs
96 – 720 hours	1.39E-04	N/A	Not modeled in current AORs
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	4.83E-04	N/A	Not modeled in current AORs
2 – 8 hours	2.58E-04	N/A	Not modeled in current AORs
8 – 24 hours	9.63E-05	N/A	Not modeled in current AORs
24 – 96 hours	6.45E-05	N/A	Not modeled in current AORs
96 – 720 hours	4.89E-05	N/A	Not modeled in current AORs

Note: (1) The control room unfiltered inleakage continues to be associated with the normal mode χ/Q of 2.55E-02 sec/m^3 until completion of control room isolation from a safety injection signal at 10 minutes.

Table 4.3-12 Assumptions Used for LOCA Analysis

	AST		CLB		Reason for Change
Core activity (containment leakage, ECCS leakage, and RWST back-leakage)	See Table 4.3-1a		See Table 4.3-1b		Updated calculations
RCS activity (containment purge)	See Table 4.3-1a		See Table 4.3-1b		Updated calculations
Fuel release fractions and timing			100% of the core activity is released immediately following event initiation		Modeling change for AST
Nuclide Group	Gap Release Phase Fraction	Early In-Vessel Phase Fraction	Gap Release Phase Fraction	Early In-Vessel Phase Fraction	
Noble gases	0.05	0.95	N/A	N/A	Modeling change for AST
Iodines	0.05	0.35	N/A	N/A	Modeling change for AST
Alkali metals	0.05	0.25	N/A	N/A	Modeling change for AST
Tellurium metals	0.00	0.05	N/A	N/A	Modeling change for AST
Barium and strontium	0.00	0.02	N/A	N/A	Modeling change for AST
Noble metals	0.00	0.0025	N/A	N/A	Modeling change for AST
Cerium	0.00	0.0005	N/A	N/A	Modeling change for AST
Lanthanides	0.00	0.0002	N/A	N/A	Modeling change for AST

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST		**CLB**		**Reason for Change**
Duration of phases					
Phase	Onset	Duration	Onset	Duration	
Gap release	30 sec	0.49167 hr	N/A	N/A	Modeling change for AST
Early in-vessel	0.5 hr	1.3 hr	N/A	N/A	Modeling change for AST
SI signal (sec)	0		N/A		Modeling update (not AST specific)
Time of control room isolation (including delays) (sec)	120		0		Modeling update (not AST specific)
Containment Leakage					
Iodine chemical form of releases (%)					
Elemental	4.85		91		Modeling change for AST
Organic	0.15		4		Modeling change for AST
Particulate	95		5		Modeling change for AST
Containment volume, maximum (ft^3)	2.7E+06		2.5E+06		Since a maximum volume is conservative, the larger value listed in USAR Section 6.2.1.5.3 was modeled
% Sprayed	85		85		No change
% Unsprayed	15		15		No change

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB	Reason for Change
Mixing between sprayed and unsprayed containment volumes (cfm)	6.94E+04	8.50E+04	The CLB value corresponds to two hydrogen mixing fans operating in slow speed. While a mixing rate of 85,000 cfm will still be present following the event, the value has been conservatively reduced to the flow rate from one containment cooler fan
Start of fan cooler mixing (min)	2	N/A	Modeling update (not AST specific)
Containment leak rates (weight %/day)			
0 – 24 hours	0.2	0.2	No change
1 – 30 days	0.1	0.1	No change
Spray timing			
Initiation (min)	2	0	The time to credit containment spray was conservatively increased to account for the delay to reaching full flow conditions
Termination (hr)	5	9.55	Conservative time used
Spray removal coefficients			
Organic iodine spray removal coefficient (hr^{-1})	0.0	0.0	No change
Elemental iodine spray removal coefficient calculations			
Spray removal coefficient (hr^{-1}), DF < 200	10	10	No change
Gas phase mass transfer coefficient (m/min)	3	N/A	Modeling update (not AST specific)
Time of fall of the spray drops (min)	0.146	N/A	Updated calculations
Volume flow rate of sprays (m^3/hr)	658.66	711.13	Conservative low flow modeled.

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB	Reason for Change
Containment sprayed volume (m^3)	6.5E+04	6.0E+04	The maximum containment volume increased from 2.5E+06 ft^3 to 2.7E+06 ft^3. Since 85% of the containment is sprayed, the sprayed volume increased proportionally with the increase in total volume.
Mass-mean diameter of the spray drops (m)	0.00116	N/A	Updated calculations
Particulate spray removal coefficient calculations			
Spray removal coefficient (hr^{-1}), DF < 50	5	0.45 (Used DFs up to 100)	Updated calculations
Spray removal coefficient (hr^{-1}), DF > 50	0.5	N/A	Updated calculations
Drop fall height (m)	35.966	36.017	Conservative rounding
Volume flow rate of sprays (m^3/hr)	658.66	711.13	Conservative low flow modeled.
Containment sprayed volume (m^3)	6.5E+04	6.0E+04	WCGS input change The maximum containment volume increased from 2.5E+06 ft^3 to 2.7E+06 ft^3. Since 85% of the containment is sprayed, the sprayed volume increased proportionally with the increase in total volume.
Ratio of dimensionless collection efficiency to average spray drop diameter			
Prior to DF of 50 (m^{-1})	10	N/A	Modeling update (not AST specific)
After DF of 50 (m^{-1})	1	N/A	Modeling update (not AST specific)

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB	Reason for Change
Particulate sedimentation removal coefficient (hr^{-1}), DF < 1000	0.1	N/A	Modeling update (not AST specific)
pH of sump	≥ 7.0	≥ 8.5	The pH value was changed to 7.0 to be consistent with the iodine retention assumptions contained within the analysis.
Control room atmospheric dispersion factors (sec/m^3)			
0 – 0.0333 hours	2.11E-03	5.30E-04	Updated calculations
0.0333 – 2 hours	6.12E-04	5.30E-04	Updated calculations
2 – 8 hours	4.38E-04	5.30E-04	Updated calculations
8 – 24 hours	1.81E-04	3.6E-04	Updated calculations
24 – 96 hours	1.29E-04	6.60E-05	Updated calculations
96 – 720 hours	9.65E-05	0	Updated calculations
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	3.91E-04	2.2E-04	Updated calculations
2 – 8 hours	2.66E-04	2.2E-04	Updated calculations
8 – 24 hours	9.62E-05	1.17E-04	Updated calculations
24 – 96 hours	7.05E-05	2.04E-05	Updated calculations
96 – 720 hours	5.52E-05	0	Updated calculations
<u>ECCS Leakage</u>			
Iodine chemical form of releases (%)			
Elemental	97	N/A	Regulatory Guide Update
Organic	3	N/A	Regulatory Guide Update
Particulate	0	N/A	Regulatory Guide Update
Sump volume (gal)	4.60E+05	4.60E+05	No change
Time to initiate ECCS recirculation (min)	0	28.2	Modeling update (not AST specific)
ECCS leakage to auxiliary building (gpm)	2	2	No change
Iodine airborne fraction	0.10	0.10	No change

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB	Reason for Change
Auxiliary building exhaust filter efficiency (all forms of iodine) (%)	90	90	No change
Control room atmospheric dispersion factors (sec/m^3)			
0 – 0.0333 hours	2.11E-03	1.10E-04	Updated calculations
0.0333 – 2 hours	6.12E-04	1.10E-04	Updated calculations
2 – 8 hours	4.38E-04	1.10E-04	Updated calculations
8 – 24 hours	1.79E-04	6.80E-05	Updated calculations
24 – 96 hours	1.14E-04	1.70E-05	Updated calculations
96 – 720 hours	8.94E-05	0	Updated calculations
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	2.80E-04	2.2E-04	Updated calculations
2 – 8 hours	1.80E-04	2.2E-04	Updated calculations
8 – 24 hours	6.44E-05	1.17E-04	Updated calculations
24 – 96 hours	4.42E-05	2.04E-05	Updated calculations
96 – 720 hours	3.22E-05	0	Updated calculations
RWST Back-Leakage			
RWST initial gas volume, minimum (gal)	3.54E+05	N/A	Modeling update
Time to initiate ECCS recirculation (min)	0	28.2	Modeling update
ECCS leakage to RWST (gpm)	3.8	3.8	No change
Iodine airborne fraction	0.10	0.10	No change
Release from RWST gas space (gpm)	3.8	3.8	No change
Iodine chemical form of releases (%)			
Elemental	97	91	Regulatory Guide Update
Organic	3	4	Regulatory Guide Update
Particulate	0	5	Regulatory Guide Update

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB	Reason for Change
Control room atmospheric dispersion factors (sec/m^3)			
0 – 0.0333 hours	1.03E-03	1.10E-04	Updated calculations
0.0333 – 2 hours	6.80E-04	1.10E-04	Updated calculations
2 – 8 hours	6.19E-04	1.10E-04	Updated calculations
8 – 24 hours	2.27E-04	6.80E-05	Updated calculations
24 – 96 hours	1.96E-04	1.70E-05	Updated calculations
96 – 720 hours	1.53E-04	0	Updated calculations
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	1.87E-04	2.2E-04	Updated calculations
2 – 8 hours	1.25E-04	2.2E-04	Updated calculations
8 – 24 hours	4.51E-05	1.17E-04	Updated calculations
24 – 96 hours	3.33E-05	2.04E-05	Updated calculations
96 – 720 hours	2.61E-05	0	Updated calculations
Containment Purge			
RCS activity released (%)	100	100	No change
Iodine chemical form of releases (%)			
Elemental	97	91	Regulatory Guide Update
Organic	3	4	Regulatory Guide Update
Particulate	0	5	Regulatory Guide Update
RCS mass, maximum (lbm)	8.42E+05	4.94E+05 CLB does not use maximum and minimum	Input change (maximum vs nominal)
Containment volume, minimum (ft^3)	2.5E+06	2.5E+06	No change
Maximum purge flow rate, unfiltered (cfm)	4,680	4,680	No change
Duration of purge release (sec)	10	8	Updated input

Table 4.3-12 Assumptions Used for LOCA Analysis (cont.)

	AST	CLB	Reason for Change
Control room atmospheric dispersion factors (sec/m^3)			
0 – 0.0333 hours	2.11E-03	1.10E-04	Updated calculations
0.0333 – 2 hours	6.12E-04	1.10E-04	Updated calculations
2 – 8 hours	4.38E-04	1.10E-04	Updated calculations
8 – 24 hours	1.79E-04	6.80E-05	Updated calculations
24 – 96 hours	1.14E-04	6.60E-05	Updated calculations
96 – 720 hours	8.94E-05	0	Updated calculations
TSC atmospheric dispersion factors (sec/m^3)			
0 – 2 hours	2.80E-04	2.2E-04	Updated calculations
2 – 8 hours	1.80E-04	2.2E-04	Updated calculations
8 – 24 hours	6.44E-05	1.17E-04	Updated calculations
24 – 96 hours	4.42E-05	2.04E-05	Updated calculations
96 – 720 hours	3.22E-05	0	Updated calculations

Table 4.3-13 Assumptions Used for Waste Gas Decay Tank Failure Analysis			
	AST	**CLB**	**Reason for Change**
Activity in ruptured tank	See Table 4.3-2a	See Table 4.3-2b	Updated calculations
Iodine chemical form of releases (%)			
Elemental	100	N/A	Modeling update (not AST specific)
Duration of release (hr)	2	2	No change
Control room isolation	None	N/A	No change
Control room atmospheric dispersion factor (sec/m^3)			
0 – 2 hours	7.17E-04	5.30E-04	Updated calculations
TSC atmospheric dispersion factor (sec/m^3)			
0 – 2 hours	1.97E-04	N/A	Not modeled in current AORs

Table 4.3-14 Assumptions Used for Liquid Waste Tank Failure Analysis			
	AST	**CLB**	**Reason for Change**
Activity in ruptured tank			
Recycle holdup tank	See Table 4.3-2a	See Table 4.3-2b	Updated calculations
Hypothetical tank maximizing iodine	See Table 4.3-2a	N/A	Updated calculations
Iodine chemical form of releases (%)			
Elemental	100	N/A	Modeling update (not AST specific)
Duration of release (hr)	2	2	No change
Control room isolation	None	N/A	No change
Control room atmospheric dispersion factor (sec/m^3)			
0 – 2 hours	7.17E-04	N/A	Not modeled in current AORs
TSC atmospheric dispersion factor (sec/m^3)			
0 – 2 hours	1.97E-04	N/A	Not modeled in current AORs

Table 4.3-15 Assumptions Used for Fuel Handling Accident Analysis			
	AST	**CLB**	**Reason for Change**
Core activity for one assembly at minimum time prior to fuel movement (Ci)			
Kr-85m	1.10E+00	See Table 4.3-4c	Updated calculations
Kr-85	5.69E+03	See Table 4.3-4c	Updated calculations
Xe-131m	5.38E+03	See Table 4.3-4c	Updated calculations
Xe-133m	1.76E+04	See Table 4.3-4c	Updated calculations
Xe-133	8.19E+05	See Table 4.3-4c	Updated calculations
Xe-135m	5.58E+01	See Table 4.3-4c	Updated calculations
Xe-135	8.14E+03	See Table 4.3-4c	Updated calculations
I-130	1.45E+02	See Table 4.3-4c	Updated calculations
I-131	4.12E+05	See Table 4.3-4c	Updated calculations
I-132	3.95E+05	See Table 4.3-4c	Updated calculations
I-133	8.90E+04	See Table 4.3-4c	Updated calculations
I-135	3.42E+02	See Table 4.3-4c	Updated calculations
Number of fuel assemblies damaged	1.2	1.2	No change
Peaking factor	1.65	1.65	No change
Time of Control room isolation (including delays) (sec)	120	N/A	No change
Decay time prior to fuel movement, minimum (hr)	76	76	No change
Gap fractions			
I-131	0.12	0.12	No change
Kr-85	0.30	0.30	No change
Other iodines and noble gases	0.10	0.10	No change
Iodine chemical form in gap (%)			
Elemental	99.85	N/A	Modeling change for AST
Organic	0.15	N/A	Modeling change for AST
Fuel pool water depth, minimum (ft)	23	23	No change
Fuel rod internal pressure, maximum (psig)	1500	1200	Modeling update (not AST specific)

Table 4.3-15 Assumptions Used for Fuel Handling Accident Analysis (cont.)

	AST	CLB	Reason for Change
Overall pool iodine DF	200	100	Regulatory Guide Update
Iodine airborne fractions (%)			
Elemental	70	N/A	Modeling change for AST
Organic	30	N/A	Modeling change for AST
Duration of release (hr)	2	2	No change
Removal of airborne activity in containment/fuel building (other than decay)	None	None	No change
Control room atmospheric dispersion factor (sec/m^3)			
Auxiliary Building Release			
HVAC Flows Except for Control Room Unfiltered Inleakage			
0 – 0.0333 hours	2.11E-03	5.30E-04	Updated calculations
0.0333 – 2 hours	6.12E-04	5.30E-04	Updated calculations
Control Room Unfiltered Inleakage			
0 – 2 hours	2.11E-03	5.30E-04	Updated calculations
Containment Release(1)			
0 – 2 hours	1.38E-03	N/A	Not modeled in current AORs
TSC atmospheric dispersion factor ($(sec/m^3)^2$)			
0 – 2 hours	3.91E-04	N/A	Not modeled in current AORs

Note: (1) The containment release χ/Q is lower that the auxiliary building release χ/Q; therefore the auxiliary building release χ/Q is used in the calculation of the doses. The containment release χ/Q is only used to determine the time the control room isolation setpoint is reached since it is lower than the containment leakage χ/Q and results in a conservative isolation time.

Table 4.3-16 Technical Support Center Parameters

	AST	CLB	Reason for Change
TSC volume (ft^3)	44,000	52,800	The CLB value is based on a simple bounding one volume calculation (length x width x height). The more accurate AST value was calculated by determining the volume of all individual TSC rooms and then summing them.
Normal ventilation flow rates (cfm)			
Unfiltered makeup flow rate	550	550	No change
Unfiltered inleakage	20	20	No change
Emergency mode flow rates (cfm)			
Filtered makeup flow rate	550	550	No change
Unfiltered makeup flow rate	0	N/A	No change
Unfiltered inleakage	20	20	No change
Filtered recirculation flow	450	450	No change
Filter efficiencies (%)			
Elemental iodine	95	90	Test procedure revised to increase acceptance criterion to 95%
Organic iodine	95	90	Test procedure revised to increase acceptance criterion to 95%
Particulates	95	90	Test procedure revised to increase acceptance criterion to 95%
Delay to switch to emergency mode of operation after event initiation (minutes)	60	0	A delay of 60 minutes was added to ensure the analysis bounds the plant specific time to switch to the emergency mode of operation after event initiation
TSC breathing rate for duration of the event (m^3/sec)	3.5E-04	3.5E-04	No change
TSC occupancy factors			
0 – 24 hours	1.0	1.0	No change
1 – 4 days	0.6	0.6	No change
4 – 30 days	0.4	0.4	No change

ARCB-RAI-2:

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

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

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

Response:

The analyses presented in Section 4.2 of Enclosure VI include a fuel management multiplier on the core inventory to accommodate anticipated changes in the values of fuel enrichment and fuel burnup, since the fuel cycle designs vary from cycle to cycle.

The fuel management multiplier is equal to [ ]^{a,c}. The fuel management multiplier increases the core activity calculated for the cycle 19 fuel cycle design, which approximates an equilibrium fuel cycle, to account for any differences in cycle 20 and future fuel cycles; therefore, it adds margin to the calculated doses. The addition of this margin is not required by Regulatory Guide 1.183 and provides an additional conservatism in the dose calculations.

The multiplier is derived from a number of historical sensitivity analyses on core activity and post-accident dose consequence analyses. Variations in fuel enrichment, core loading (i.e. fuel mass), and cycle burnup were considered in the sensitivities analyses. Based on these sensitivity analyses, a value of [ ]^{a,c} was used for the WCGS.

ARCB-RAI-3

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

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

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

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

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

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

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

Response:

a) Regulatory Position 1.3.2 of Regulatory Guide 1.183 specifies that an evaluation of AST impact was performed on three representative operating reactors and concluded that the Technical Information Document (TID)-14844 based methodology and assumptions generally bound the AST methodology. Regulatory Position 1.3.2, states, in part:

    This evaluation determined that radiological analysis results based on TID-14844 source term assumptions…and the whole body and thyroid methodology generally bound the results from the analyses based on AST and TEDE methodology. Licensees may use the applicable conclusions of this evaluation in addressing the impact of the AST on design basis radiological analyses.

   Full implementation of AST at WCGS does not impact the assumptions or inputs to the current TID based analyses. As such, AST implementation would not impact the conclusions from Regulatory Position 1.3.2, the TID based analyses and results, or the ability to implement AST in other radiological analyses per the guidance in Regulatory Position 1.3.4.

b) No new operator actions have been credited to support the AST methodology. Regarding the time critical action to isolate the failed CREVS train, the overall action

described in USAR Section 15A.3.1 remains unchanged, i.e., operator action is required to ensure no bypass pathways exist for unfiltered air to enter the control room following the failure of a filtration fan. However, the damper modification being implemented will eliminate the possibility of having to locally isolate the failed CREVS train. Specifically, the modification will result in the system automatically isolating the CREVS by normal system operation when the control room AC fan unit is turned off, which can be performed remotely from the control room following the identification of the failed filtration fan.

c) The requirements for the post-accident sampling system were eliminated in Amendment No. 137 (Reference 5) and thus is unaffected by the change to AST methodology. With regards to the containment high range radiation monitors, the controlling isotope for the TID-14844 source term, Xe_{133}, remains the controlling isotope for the AST source term. Thus, the containment high range radiation monitors are also unaffected.

d) WCNOC is only proposing exceptions to the full scope implementation of the AST for equipment qualification and NUREG-0737 evaluations other than Control Room Habitability Envelope (CRHE) and Technical Support Center (TSC) doses. With regards to NUREG-0737 evaluations, full implementation of AST at WCGS does not impact the assumptions or inputs to the current TID based analyses, as documented in the part a) response to this request. Thus, AST implementation would not impact their conclusions from Regulatory Position 1.3.2, the TID based analyses, their results, or the ability to implement AST in other radiological analyses per the guidance in Regulatory Position 1.3.4. Once AST is part of the site's licensing basis this regulatory position specifies that "…all characteristics of the AST and the TEDE criteria incorporated into the design basis will be addressed in all affected analyses on an individual as-needed basis." WCNOC plans to incorporate AST into these remaining analyses on an as needed basis as described in Regulatory Position 1.3.4.

ARCB-RAI-4:

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

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

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

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

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

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

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

Response:

a) The core is modeled in seven fuel regions (batches).

b) The length of cycle 19 was modelled as 511 effective full-power days (EFPD), which bounds the actual cycle length of 505 EFPD. Prior cycles were assumed to operate for 455 EFPD consistent with the WCGS fuel cycle operating experience. The region-specific data is tabulated below for cycle 19.

a,c

Region	# of FAs	Wt% U-235	MWt/FA (pre- cycle 19)	MWt/FA (cycle 19)

c) The activity reported in Enclosure IV, Table 4.2-1 is taken at the end of cycle 19 for each fuel region. (511 EFPD)

d) The LOCA dose consequence analysis modeled the 63 nuclides presented in Enclosure IV, Table 4.3-1a. The main steamline break, loss of non-emergency AC power, locked rotor, rod ejection, letdown line break, and steam generator tube rupture dose consequence analyses consider the 21 noble gas, iodine, and alkali metal nuclides listed in Enclosure IV, Table 4.3-1a. The fuel handling accident dose consequence analysis (both inside containment and in the fuel building) considers the 12 noble gas and iodine nuclides listed in Enclosure IV, Table 4.3-15. The tank rupture dose consequence events (waste gas decay tank failure and liquid waste tank failure) consider the 16 noble gas and iodine nuclides listed in Enclosure IV, Table 4.3-2a.

e) The maximum enrichment considered for a single fuel assembly is [ ][a,c] wt% U-235, consistent with the cycle 19 fuel cycle design.

In general, radionuclides with short half-lives are present in quantities that are proportional to core power. Those are conservatively treated by including the emergency core cooling system (ECCS) evaluation uncertainty in core thermal power. Quantities of longer-lived radionuclides will build up during the fuel cycle. Those are conservatively treated by modelling the longest anticipated cycle 19 length (511 EFPD). The end of life values are reported in Enclosure IV, Table 4.2-1.

This is consistent with Regulatory Guide 1.183 Section C.3.1, which states:

> "The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[9]
>
> [9] Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used."

ARCB-RAI-5

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

Response:

The fuel handling accident (FHA) analysis conservatively assumed that 100% of the fuel will not meet the Footnote 11 from Regulatory Position 3.2 of Regulatory Guide 1.183. Footnote 11 contains burnup and linear heat generation rate limits for the gap fractions contained in Table 3 of Regulatory Guide 1.183 as well as an alternative method for determining gap fractions provided they bound the limiting projected plant-specific power history for the specific load. The analysis conforms with the position such that alternative gap fractions were used which were appropriate for the assumption of not meeting the Footnote 11 limits. These gap fractions are obtained from Regulatory Guide 1.25, as modified by NUREG/CR-5009, which provides higher, i.e., more conservative gap fractions than Regulatory Guide 1.183, which are not constrained by the Footnote 11 burnup limits as they can be applied to higher burnup to bound power history.

ARCB-RAI-6:

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

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

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

Response:

The sprayed and unsprayed regions of containment were modeled as separate compartments. Activity removal via natural deposition was only modeled in the unsprayed region of containment and in the sprayed region of containment when sprays were not credited for removal of aerosols (e.g., after containment spray termination). Activity removal via the sprays and activity removal via natural deposition were not modeled concurrently in the same compartment of containment. Therefore, there was no double-accounting for the removal of aerosols in either the sprayed or unsprayed regions of containment. The user-defined coefficients in the RADTRAD model for sprays and natural deposition were used.

There is no other credit for reduction of aerosols in any pathway except for removal by filters in the control building/control room ventilation systems. This removal does not

compete with any other removal process and is justified due to the availability of the filters in these systems.

The assumed sedimentation rate does impact the final dose results.

<u>ARCB-RAI-7:</u>

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

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

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

Response:

In accordance with Standard Review Plan (SRP) Section 6.5.2, the spray removal coefficient, if calculated to be greater than 20 hr^{-1}, is to be limited to 20 hr^{-1}. For the analysis performed for the WCGS, the removal coefficient is calculated to be greater than 20 hr^{-1}; however, the removal coefficient is reduced further to 10 hr^{-1} to conservatively address iodine loading in the spray fluid during recirculation. During recirculation, the spray solution will gradually become loaded with elemental iodine, which will limit the capacity of the spray to remove airborne iodine. The spray removal coefficient would be inversely proportional to the decontamination factor achieved for elemental iodine. Thus, when recirculation spray is first credited, there is a low enough level of elemental iodine in the sump solution that it is appropriate to use the removal coefficient of 20 hr^{-1}. However, when the decontamination factor approaches its defined limit of 200, the removal coefficient would be only a small fraction of its original value. The impact of this varying nature of the removal coefficient is approximated by setting the removal coefficient to one half of the calculated value. However, for conservatism and simplification, a value of 10 hr^{-1} (half of the 20 hr^{-1} limits specified in SRP 6.5.2) was used.

The spray removal coefficient is a function of the spray flow rate. The analysis used a spray removal coefficient based on a conservatively low spray flow rate (658.66 m^3/hr) to bound both modes (injection and recirculation, with spray flow rates of 665.86 m^3/hr and 780.00 m^3/hr, respectively) of spray operation. Therefore, sensitivities on timing on the spray switchover times are not necessary.

The lower spray flow rate and the factor of 2 reduction in the removal coefficient applied for the entire duration of spray operation bounds the impacts on the analysis results due to potential changes in switchover timing.

The following wording is added to Enclosure IV, Section 4.3.9.2.2.1:

"During recirculation, the spray solution will gradually become loaded with elemental iodine, which will limit the capacity of the spray to remove airborne iodine. The spray removal coefficient would be inversely proportional to the decontamination factor (DF) achieved for elemental iodine. Thus, when recirculation spray is first credited, there is a low enough level of elemental iodine in the sump solution that it is appropriate to use the removal coefficient of 20 hr^{-1}. However, when the DF approaches its defined limit of 200, the removal coefficient would be only a small fraction of its original value. The impact of this varying nature of the removal coefficient is approximated by setting the removal coefficient to one half of the calculated value.

Also, the spray removal coefficient is a function of the spray flow rate. The analysis used a spray removal coefficient based on a conservatively low spray flow rate to bound both modes (injection and recirculation) of spray operation."

<u>ARCB-RAI-8:</u>

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

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

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

Response:

As leakage enters the RWST at a rate of 3.8 gpm (assumed to be all liquid), the gas volume that can be credited for dilution (initially 3.54E+05 gallons) is reduced, resulting in an increased concentration of activity in the RWST gas volume. The leakage is modeled for the duration of the analysis, resulting in a gas volume of 189,840 gallons at 720 hours. Because RADTRAD maintains a constant compartment volume, the effects of the increased concentration are accounted for via an increased flow rate from the RWST. The flow rate is increased in proportion to the expected increase in concentration resulting from the reduced RWST gas volume.

The following wording is added to Enclosure IV, Section 4.3.9.2.2.3:

"The RWST gas volume is decreased by the rate of leakage into the RWST."

ARCB-RAI-9:

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

Response:

The following summarizes the doses for each pathway analyzed for the LOCA dose consequence analysis:

Exclusion Area Boundary (EAB) (0.5 hr to 2.5 hr):

Containment Leakage	4.73E+0 rem TEDE
ECCS Leakage	4.63E-1 rem TEDE
RWST Back-Leakage	4.36E-3 rem TEDE
Containment Purge	0.00E+0 rem TEDE

Low Population Zone (LPZ):

Containment Leakage	2.00E+0 rem TEDE
ECCS Leakage	1.26E+0 rem TEDE
RWST Back-Leakage	1.57E+0 rem TEDE
Containment Purge	8.24E-4 rem TEDE

Control Room:

Containment Leakage	1.01E+0 rem TEDE
ECCS Leakage	8.48E-1 rem TEDE
RWST Back-Leakage	2.69E+0 rem TEDE
Containment Purge	6.52E-2 rem TEDE
External Sources	1.53E-1 rem TEDE

The worst 2-hour EAB dose was determined by summing all the time-dependent dose pathways and then determining the worst 2-hour dose. Note that releases via the mini-purge terminate prior to 0.5 hours and thus do not contribute to the limiting 2-hour EAB dose.

This information is included in Enclosure IV, Section 4.3.9.4.

ARCB-RAI-10:

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

> *The FIPCO-V computer code calculates the buildup of fission product activities in plant systems and components, including the reactor coolant system, chemical and volume control*

system demineralizer resins, VCT [volume control tank] liquid and vapor phases, and waste decay tank (WGDT).

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

Response:

The equations which are used to determine the concentrations of fission products at any time are developed from the general balance equation:

$$\begin{bmatrix} \text{Net Rate of} \\ \text{Accumulation} \end{bmatrix} = \begin{bmatrix} \text{Rate of} \\ \text{Formation} \end{bmatrix} - \begin{bmatrix} \text{Rate of} \\ \text{Loss} \end{bmatrix}$$

<u>Reactor Core Fission Products</u>

The buildup of fission products in the fuel assemblies in the core is described by the following equations:

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

[

]a,c

The inputs used in the FIPCO calculations are provided in the following two tables. The data below are taken from operating data, fuel cycle design data, and plant information.

Key Input Parameters for WCGS RCS Activity Analyses

a,c

Parameter	Value	Notes
Core thermal power, MWt		
Fuel Management multiplier		
RCS boron concentrations		
Fuel defect level, %		
RCS volume, ft^3		
RCS expansion factor		
Steam generator (S/G) tube plugging level, %		
Normal pressurizer liquid level @ full power, %		
Core T_{avg}, ^{o}F		
Letdown flow rate, gpm		
Volume control tank (VCT) volume, gallons/ft^3		
VCT liquid volume, %		
VCT conditions: Temperature, ^{o}F Pressure, psig		
VCT continuous purge rate, scfm		.
Gas decay tank volume, ft^3		
Number of gas decay tanks		.
Effective deborating demineralizer cut-in concentration, ppm		
Mixed bed resin volume, ft^3		

Critical Boron Concentrations

a,c

Burnup (MWd/MTU)	B Conc. (ppm)	Burnup (MWd/MTU)	B Conc. (ppm)	Burnup (MWd/MTU)	B Conc. (ppm)

<u>ARCB-RAI-11:</u>

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

Response:

The forms of radioiodine assumed for the containment purge pathway is 97% elemental and 3% organic. This is an assumed split consistent with other releases specified in Regulatory Guide 1.183. There is no guidance on forms of radioiodine for purge releases in Regulatory Guide 1.183 and the split has no impact on the analysis results since no spray removal is credited for the RCS activity in containment. The Control Room Emergency Ventilation System (which includes the Control Building) filters are modeled with the same efficiency for all forms of iodine.

The following wording is added to Enclosure IV, Section 4.3.9.2.2.4:

“Note that the assumed iodine chemical fractions do not impact the analysis results since spray removal is not credited for the RCS activity in containment.”

The following wording is added to Enclosure IV, Section 4.3.9.2.3:

“Note that the assumed iodine chemical fractions for the containment purge pathway do not impact the analysis results since the emergency mode ventilation system filters are modeled with the same efficiency for all forms of iodine.”

ARCB-RAI-12

NRC Information Notice (IN) 2012-01: "Seismic Considerations- Principally Issues Involving Tanks," dated January 26, 2012 (ADAMS Accession No. ML 11292A175), provides examples and references to events in which licensees failed to recognize various seismic considerations and system alignment issues that could impact safety. The NRC staff has identified recent concerns about licensees who failed to recognize that aligning non-seismic piping to the RWST would require TS limiting condition for operation (LCO) action statement entry, system modifications, or license amendments.

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

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

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

Response:

WCNOC review of NRC Information Notice 2012-01 and associated operating experience is documented in the corrective action program. The Refueling Water Storage Tank (RWST) is described in USAR Section 6.3 as part of the Emergency Core Cooling System (ECCS) and its ECCS function is to contain an inventory of 394,000 gallons of water to provide for ECCS mitigation purposes. The RWST is required to perform its function during the short term (less than 24 hours) following a LOCA, MSLB, SGTR or any other accident requiring the ECCS following receipt of an safety injection (SI) signal. The ECCS consists of three separate subsystems: centrifugal charging (high head), SI (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains.

A review of the interface piping connections with the RWST was performed. There are eight interface piping connections to the seismic category 1 piping of the RWST excluding the piping connections from RWST supply header to the suctions of the ECCS pumps and Containment Spray (CS) System pumps. The interface piping connections from RWST supply header to the suctions of the ECCS pumps and CS System pumps are seismic category 1 piping connections.

The remaining eight interface piping connections on the RWST were reviewed for seismic to non-seismic piping alignment concerns as follows:

1. RWST to Spent Fuel Pool Cooling and Cleanup System Flow Path

The Spent Fuel Cooling and Cleanup System consists of three subsystems: fuel pool cooling subsystem, fuel pool cleanup subsystem and fuel pool surface skimmer subsystem. The fuel pool cleanup subsystem is non-safety related and has no safety design basis. The fuel pool cleanup subsystem limits the fission and corrosion product concentrations in the refueling pool water, the transfer canal water, and the fuel storage pool water to permit operator access to the fuel storage area and for fuel handling operations.

WCGS has two safety related air operated isolation valves (BNHV8800A/B) in series on the safety related piping from the RWST before it is connected to the non-safety related piping of the fuel pool cleanup subsystem. By design, valves BNHV8800A/B receive a safety injection signal to initiate automatic closure. The valves receive independent and redundant auto closure signals (i.e., each valve is actuated from a different SI actuation train) ensuring these safety related isolation valves are closed when the RWST may be required to support the ECCS. The fail-safe position of these valves is “closed” (the air solenoid fails open on a loss of power). The design features of the interconnecting piping between the RWST and the fuel pool cleanup subsystem meet GDC 35, “Emergency core cooling.” This criterion is satisfied by these two valves in that the system safety function can be accomplished with the failure of one valve as the other valve will fulfill the isolation function. Following a design basis accident that requires the ECCS mitigation function, a safety injection signal is generated and results in the closure of BNHV8800A and BNHV8800B, if the RWST was in the purification alignment mode.

2. Safety Injection (SI) Pump Recirculation Flow Path to RWST

The SI pumps are provided with a recirculation flow path to the top of the RWST to support pump operation. The SI pump’s recirculation flow path to the RWST is not in use during normal plant operations as the SI pumps are in standby for ECCS mitigation. The SI pump’s recirculation flow path is not used to bring the plant to a cold shutdown condition. The SI pump’s recirculation flow path is typically used to support pump operation for pump testing during normal plant operations. The SI pump recirculation flow path to the RWST is seismically qualified piping (seismic category 1 and II/I Special Scope). II/I Special Scope (SS), is non-safety related piping that runs above safety related piping that has been supported in this area seismically so as not to degrade the function of the safety related piping below it. Piping with this qualification is designed and constructed to ensure that its failure will not reduce the functioning of a safety related component to an unacceptable safety level.

3. Containment Spray (CS) Pump Test Flow Path to RWST

The CS pumps are provided with a flow path to the top of the RWST to provide a recirculation flow path to test each pump or to support system testing. The CS pump's recirculation flow path to the RWST is seismically qualified piping (seismic category 1 and II/I SS).

4. Charging and Volume Control System (CVCS) Boric Acid Blending Tee Flow Path to RWST

The CVCS boric acid blending tee flow path to the top of the RWST provides for the addition of blended boric acid solution to the RWST or a path for testing the boric acid transfer pumps. The CVCS boric acid blending tee flow path to the RWST is seismically qualified piping (seismic category 1 and II/I SS).

5. SI System Test Flow Path to RWST

The SI System test flow path to the top of the RWST allows for the testing of ECCS check valves, supports depressurization of ECCS piping, and provides dampening of the RHR and SI pump start pressure transients. The non-safety related piping in the SI System test flow path is ¾ inch piping. Due to this small line size, a break or crack in this size of line is not postulated and isolation is provided by the containment isolation valves.

6. RHR Pump Flow Path to RWST

The RHR Pump flow path to the RWST is only used during refueling outages to transfer/drain the refueling pool water to the RWST. An isolation valve (BN8717) is normally locked closed and is opened under administrative controls when the plant is in a refueling outage. The RHR pump flow path to the RWST is seismic category 1 qualified piping.

7. RWST Drain Flow Path to Waste Holdup Tank

The RWST can be drained via a flow path to the Waste Holdup Tank in the Radwaste Building. This drain path is seismic category 1 qualified piping out to and including its isolation valve BNV0017 that is locked closed. Downstream of the isolation valve, the piping is classified as non-safety related. When the RWST is required to be drained for maintenance or internal inspection during refueling outages, this drain flow path is utilized.

8. RWST Overflow Flow Path to Waste Holdup Tank

The RWST is provided with an overflow path at the top of the RWST that is also connected to the Waste Holdup Tank. The overflow piping is seismic category 1 qualified near the RWST and connects with non-safety related piping going to the Waste Holdup Tanks. There are no isolation valve(s) in the overflow flow path as it is designed to provide overflow protection for the RWST.

The piping aligned to systems which recirculate post-LOCA sump fluid is seismically qualified piping.

Relevant Details of AST Analysis Model:

The AST analysis models RWST back-leakage at a rate of 3.8 gpm from the sump. The activity is modeled to be delivered directly to the gas filled portion of the RWST; however, only 10 percent of the activity becomes airborne and is available for release to the environment. The release rate from the RWST to the environment is based on the volume displacement from the incoming leakage. An adjustment to the release rate is made to account for a reduction in the RWST gas volume available for dilution as the leakage into the RWST increases the water level. The adjusted flow rate is based on increasing the modeled flow in order to exhaust the appropriate activity concentration. This is necessary as RADTRAD maintains a constant compartment volume and transfers are concentration-based.

The RWST modeling description is included in Enclosure IV, Section 4.3.9.2.2.3 and also included as a response to ARCB-RAI-8 in Enclosures VI and VII.

ARCB-RAI-13

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

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

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

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

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

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

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

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

Response:

The R-23 detector discussed above is hereafter referred to as the control room air supply radiation monitors (GKRE0004 and GKRE0005).

a) The total time to isolate was calculated by determining a time by which the isolation signal(s) would have been generated and then adding a delay to account for the Control Room Ventilation Isolation Signal (CRVIS) to actuate the Control Room Emergency Ventilation System (CREVS). Isolation signals include safety injection or high radiation in the control room air supply radiation monitors. To ensure that the time delay modeled in the analyses is bounding, new TS Surveillance Requirement (SR) 3.3.7.6 has been added for the performance of the required response time verification.

   For the loss of coolant accident (LOCA) radiological consequences analysis, the safety injection isolation signal is assumed to occur immediately after event initiation. For conservatism, the switchover to emergency mode ventilation was not assumed to occur until 2 minutes after the onset of the event (including the 1 minute delay for switchover). This is presented in Enclosure IV of this LAR, Section 4.3.9.2.3.

   For the main steam line break (MSLB) radiological consequences analysis, the safety injection isolation signal is assumed to occur immediately after event initiation. For conservatism, the switchover to emergency mode ventilation was not assumed to occur until 90 seconds after the onset of the event (including the 1 minute delay for switchover). This is presented in Enclosure IV of this LAR, Section 4.3.3.2.3.

   For the rod ejection containment leakage case, the safety injection isolation signal is assumed to occur within 150 seconds of event initiation. The switchover to emergency mode ventilation was not assumed to occur until 210 seconds after the onset of the event (including the 1 minute delay for switchover). This is presented in Enclosure IV of this LAR, Section 4.3.6.2.3.

   For the steam generator tube rupture (SGTR), rod ejection primary-to-secondary case, locked rotor, and fuel handling accident (FHA) radiological consequences analysis, the control room air supply radiation monitors setpoint is shown to be reached for each accident immediately after the onset of the event. This is based on the amount of Xe-133 activity initially in the primary system, the environmental release rate modeled for each analysis, and the atmospheric dispersion factor modeled for the control room. Since the control room air supply radiation monitors are located in the control room HVAC ductwork, the concentration of activity entering the control room is the concentration seen by the monitors. The delay from the control room air supply

radiation monitors setpoint being reached to the switchover to emergency mode ventilation is 60 seconds. Each analysis conservatively assumed that switchover did not occur for at least 120 seconds. This is presented in Enclosure IV of this LAR, Sections 4.3.8.2.3, 4.3.6.2.3, 4.3.5.2.3, and 4.3.12.2.3.

b) For the LOCA, rod ejection containment leakage case, and MSLB analyses, a safety injection signal is credited. For LOCA and rod ejection containment leakage, the safety injection signal is generated by the reduction in RCS pressure as a result of the respective breaks. For MSLB, the safety injection signal is generated by the low steamline pressure as a result of the break. This is presented in Enclosure IV of this LAR, Sections 4.3.9.2.3, 4.3.6.2.3, and 4.3.3.2.3.

c) The CRVIS time, emergency diesel generator startup time, time to load the electrical bus and damper closure times have been considered and are bounded by the 60 second delay time assumed for each accident scenario as described in response to item a) above.

   The worst case single failure of a control room filtration fan is considered and results in a pathway for contaminated air from the control building to bypass the control room filtration system filter adsorber unit and thereby enter the control room unfiltered. The mitigative action for this single failure is an operator action to isolate this flow path by closing the appropriate train-specific damper. This single failure is the same as that assumed in the current licensing basis analysis of record (AOR), where the operator action is assumed to take place 1.5 hours after event initiation. Since the two CREVS trains are independent, this single failure only affects one train of the CREVS.

d) The control room air supply radiation monitors are credited to isolate the control room in the SGTR, rod ejection primary-to-secondary leakage, locked rotor, and FHA analyses. The activity concentration seen by the control room air supply radiation monitors setpoint is determined by the following equation:

   Activity Concentration at Control Room Air Supply Radiation Monitors = Activity Conc * Environment Release Rate * CR χ/Q

   The following calculations determine the time to reach the setpoint for each accident. Note that the analyses modeled a control room radiation monitor setpoint of 2.12E-3 Ci/m^3 Xe-133.

<u>SGTR</u>

Since the analysis assumes no delay in the release of activity and no delays in the transport of activity through the primary and secondary systems or in transport from the release point to the air intake, an instantaneous generation of the high radiation signal could be assumed. The total Xe-133 activity is 4.19E4 Ci and the RCS mass is 3.99E5 lbm. The initial break flow rate is 3085 lbm/min and the 0-2 hour CR χ/Q for steaming releases from the SG for normal operation is 2.55E-2 sec/m^3.

The activity concentration at the detector is:

$$(4.19E4 \text{ Ci} / 3.99E5 \text{ lbm})(3085 \text{ lbm/min} * 1 \text{ min}/60 \text{ sec})(2.55E\text{-}2 \text{ sec/m}^3) = 1.4E\text{-}1 \text{ Ci/m}^3$$

The control room radiation monitor is shown above to exceed its setpoint immediately.

This comparison is presented in Enclosure IV of this LAR, Section 4.3.8.2.3.

Note that Enclosure IV, Section 4.3.2.1 describes that isolation based solely on the control room radiation monitor does not fully switch the HVAC systems such that the control room unfiltered inleakage continues to be associated with the normal mode X/Qs. The SGTR analysis credits the safety injection signal due to low pressurizer pressure to complete isolation of the control room. From Enclosure I, Section 2.7.3, the safety injection signal occurs less than 6 minutes after the onset of the event. For conservatism, the isolation of the control room is completed at 10 minutes after the onset of the event.

<u>Rod Ejection Primary-to-Secondary Leakage</u>

Since the analysis assumes no delay in the release of activity from the damaged fuel and no delays in the transport of activity through the primary and secondary systems or in transport from the release point to the air intake, an instantaneous generation of the high radiation signal could be assumed. The total Xe-133 activity is 4.15E6 Ci and the RCS mass is 3.99E5 lbm. The primary to secondary leak rate is 8.342 lbm/min (applying a conservative analysis density of 62.4 lbm/ft^3) and the 0-2 hour CR χ/Q for steaming releases from the SG for normal operation is 2.55E-2 sec/m^3.

The activity concentration at the detector is:

$$(4.15E6 \text{ Ci} / 3.99E5 \text{ lbm})(8.342 \text{ lbm/min} * 1 \text{ min}/60 \text{ sec})(2.55E\text{-}2 \text{ sec/m}^3) = 3.7E\text{-}2 \text{ Ci/m}^3$$

The control room air supply radiation monitors are shown above to exceed the setpoint immediately.

This comparison is presented in Enclosure IV of this LAR, Section 4.3.6.2.3.

<u>Locked Rotor</u>

Since the analysis assumes no delay in the release of activity from the damaged fuel and no delays in the transport of activity through the primary and secondary systems or in transport from the release point to the air intake, an instantaneous generation of the high radiation signal could be assumed. The total Xe-133 activity is 8.30E5 Ci and the RCS mass is 3.99E5 lbm. The primary to secondary leak rate is 8.342 lbm/min (applying a conservative analysis density of 62.4 lbm/ft^3) and the 0-2 hour CR χ/Q for steaming releases from the SG for normal operation is 2.55E-2 sec/m^3.

The activity concentration at the detector is:

(8.30E5 Ci / 3.99E5 lbm)(8.342 lbm/min * 1 min/60 sec)(2.55E-2 sec/m^3) = 7.4E-3 Ci/m^3

The control room air supply radiation monitors are shown above to exceed the setpoint immediately.

This comparison is presented in Enclosure IV of this LAR, Section 4.3.5.2.3.

<u>FHA</u>

Since the analysis assumes no delay in the release of activity from the damaged fuel and no delays in the transport of activity through the spent fuel pool or in transport from the release point to the air intake, an instantaneous generation of the high radiation signal could be assumed. The total Xe-133 activity is 1.62E5 Ci. Given the linear release over a 2-hour period, the release rate is 8.1E4 Ci/hr or 22.5 Ci/sec. The 0-2 hour CR χ/Q for releases is 1.38E-3 sec/m^3. (Note that this is the χ/Q associated with releases from a FHA inside containment. It is selected for this check as the lower χ/Q is conservative in determining when the setpoint is reached.) Therefore the concentration of Xe-133 at the control room air supply radiation monitors is:

(22.5 Ci/sec)(1.38E-3 sec/m^3) = 3.11E-2 Ci/m^3

The control room air supply radiation monitors are shown above to exceed the setpoint immediately.

This comparison is presented in Enclosure IV of this LAR, Section 4.3.12.2.3.

e) The control room air supply radiation monitors are located in the control room HVAC ductwork.

f) Yes, the control room air supply radiation monitors comply with Regulatory Position 5.1.2 of Regulatory Guide 1.183. Specifically, compliance is demonstrated by the following:

   1) The two redundant radiation monitors and associated actuation instrumentation are safety-related.

   2) The radiation monitors are required to be operable by TS limiting condition for operation (LCO) 3.3.7, "CREVS Actuation Instrumentation."

   3) The radiation monitors are powered by Class 1E electrical equipment.

   4) The control room air supply radiation monitors provide input to initiate a CRVIS. Proper actuation of the CREVS is verified in the emergency operating procedures.

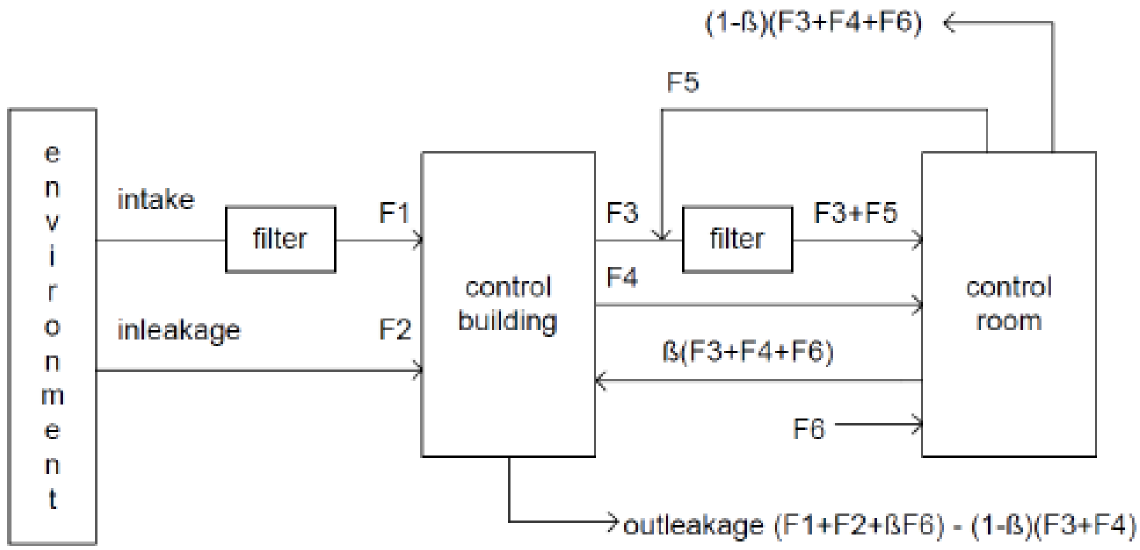
ARCB-RAI-14

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

Response:

The figure displayed below (obtained from Section 15A.3 of the Wolf Creek USAR), provides a diagram of the flow rates modeled for the control building and control room. The "Normal Mode HVAC" flow rates are modeled prior to CREVS actuation. Following the CREVS actuation, the "Emergency Mode HVAC Prior to Operator Action" provides the flow rates considering a single failure of a control room filtration fan, which is consistent with the analysis of record single failure. The "Emergency Mode HVAC after Operator Action" flow rates are modeled after the failed train has been isolated by closing the train-specific damper.

Enclosure IV, Table 4.3-5 contains the flow rates input values used in the analysis. A value of 400 cfm was modeled as the inleakage into the control building. The inleakage value was increased from the analysis of record value of 300 cfm to allow additional testing margin. An acceptance criterion of 300 cfm was confirmed in October 2010 during performance of TS SR 3.7.10.4. Although the analysis acceptable inleakage increased, the previous TS surveillance test performance in accordance with the Control Room Envelope Habitability Program (TS 5.5.18) demonstrates that the control building inleakage is less than 400 cfm. The Frequency of SR 3.7.10.4 is also in accordance with the Control Room Envelope Habitability Program. This Frequency is procedurally controlled to 6 years.



ARCB-RAI-15

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

Response:

In Reference 1, WCNOC proposed to adopt TSTF-51-A, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations." The adoption of TSTF-51-A revised TS 3.3.8 by adding "recently" to the Applicability. The revised Applicability stated: "During movement of recently irradiated fuel assemblies in the fuel building." In the response to ARCB-RAI-24, WCNOC identified that the changes proposed in TSTF-51-A, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations." are being

withdrawn. Subsequently, WCNOC withdrew the license amendment request (Reference 1) in WCNOC letter dated June 18, 2014 (Reference 3). The resubmittal of the license amendment request excluded the adoption of TSTF-51-A, Revision 2. With the removal of changes proposed by TSTF-51-A, the Emergency Exhaust System (EES) actuation instrumentation is required to be OPERABLE during movement of irradiated fuel assemblies in the fuel building. As such, if a FHA were to occur, upon receipt of a fuel building ventilation isolation signal generated by the gaseous radioactivity monitors in the fuel building exhaust line, normal air discharges from the building are terminated, the fuel building is isolated, and the stream of ventilation air discharges through the EES filter trains.

However, as described in Enclosure IV of this LAR, Section 4.3.12, while the EES is credited for the event and thus the release location will be the unit vent for a fuel handling accident within the fuel building, filtration from the EES was conservatively not credited for the FHA analysis as the dose limits were met without crediting EES filtration.

Updates to the TS Bases are presented in Enclosure IV of this LAR, page B 3.3.8-1 of Section 11.

ARCB-RAI-16

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

a) *Please justify the use of 50 cfm for unfiltered in-leakage in the configuration when the control room ventilation system is not isolated. Please state if this value has been confirmed by testing and will it be confirmed periodically as part of a TS surveillance program. (Note this issue was previously identified in NRC Regulatory Issue Summary 2006-04, Issue Number 3.)*

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

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

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

Response:

In the response to ARCB-RAI-24, WCNOC identified that the changes proposed in TSTF-51-A, Rev. 2, are being withdrawn. As such, the normal HVAC lineup is not credited to mitigate the effects of a fuel handling accident. Specific responses are provided below:

a) The 50 cfm of unfiltered inleakage in the configuration when the control room ventilation system is in the normal mode of operation is consistent with the unfiltered inleakage in the configuration when the CREVS is in service. The 50 cfm is conservatively added to the 1950 cfm air makeup flow to the control room to yield a total flow rate of 2000 cfm of unfiltered atmosphere into the control room. Since the normal makeup flow rates are not credited to mitigate the consequence of a fuel handling accident there is not a TS Surveillance Requirement to measure the normal makeup flow rates and inleakage.

b) As stated in the response to ARCB-RAI-24, WCNOC identified that the changes proposed in TSTF-51-A, Rev. 2, are being withdrawn. As such, the normal makeup for the control building and control room is not credited to mitigate the consequences of a fuel handling accident.

c) The normal makeup flow rates are no longer credited to mitigate the consequence of a fuel handling accident. Thus, there is not a TS Surveillance Requirement for the normal makeup flow rates. A sensitivity analysis was not performed for the normal flow rates. However, the actual normal makeup flow rates were measured and compared to the values modeled in the dose analyses. The normal makeup flow rates modeled in the dose analyses are greater than the measured plant flow rates for the control building and control room by more than 10%.

d) The following analyses only credit normal makeup flow rates to mitigate the consequences of the accident: loss of alternating current power, letdown line break, and tank ruptures. The assumptions used in these analyses are consistent with the assumptions for the normal mode of operation used in each accident analyzed for the AST implementation, as described in Section 4.3.2.1 and Table 4.3.5 of Enclosure IV of this LAR. Remaining in normal mode of operation is conservative for the accidents since the normal mode flow rates are significantly higher than the emergency mode flows and no filtration is credited for activity entering the control building or control room during normal mode. In addition, filtered recirculation is credited for activity removal in the control room during emergency mode of operation. Thus, the activity in the control room is greater for normal mode of operation.

   The following wording is added to Enclosure IV of this LAR, Sections 4.3.4.2.3, 4.3.7.2.3, 4.3.10.2.3 and 4.3.11.2.3: "This modeling is conservative since activity reduction due to filtration of inflow and filtered recirculation is not credited."

   A sensitivity analysis was not performed for the normal flow rates. However, the actual normal makeup flow rates were measured and compared to the values modeled in the dose analyses. The normal makeup flow rates modeled in the dose analyses are greater than the measured plant flow rates for the control building and control room by more than 10%.

<u>ARCB-RAI-17</u>:

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

Response:

A detailed discussion of the waste gas decay tank failure calculation and justification for the assumption is provided in the response to ARCB-RAI-31 below.

ARCB-RAI-18

Page 15.7-13 of the UFSAR "markup" states that the gap fractions are obtained for high burnup fuel from Regulatory Guide (RG) 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," dated March 1972 (ADAMS Accession No. ML083300022), as modified by NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," dated February 1988 (not publicly available), to support the conservative assumption that 100 percent of the rods do not meet the burnup and kilowatts per foot (kW/ft) limits set forth in Footnote 11 of RG 1.183. Enclosure VI, page 4-69 of the submittal elaborates on the use of NUREG/CR-5009.

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

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

Response:

a) The following table compares the nuclide gap fractions used in the WCGS Fuel Handling Accident analysis (which are obtained from RG 1.25, as modified by NUREG/CR-5009) versus those specified in Table 2.9 of PNNL-18212, Revision 1 (Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard, ADAMS Accession Number ML112070118), which calculated the gap release fractions utilizing ANSI/ANS-5.4-2011.

Nuclide	WCGS Analysis Gap Fractions	Calculated Gap Fractions Utilizing ANSI/ANS-5.4-2011
Kr-85	0.30	0.38
I-131	0.12	0.08
I-132	0.10	0.09
Other Noble Gases	0.10	0.08
Other Halogens	0.10	0.05

As shown in the table above, the gap fractions used in the WCGS analysis are conservatively higher than those specified in ANSI/ANS-5.4-2011, with the exception of Kr-85. Kr-85 is not a significant dose contributor when compared to the other nuclides being modeled. Kr-85 has an effective dose equivalent (EDE) dose conversion factor (DCF) of 1.19E-16 Sv-m^3/Bq-sec and a released activity of 3.38E+03 Ci, whereas Xe-133 (a major dose contributor) has an EDE DCF of 1.56E-15 Sv-m^3/Bq-sec and a released activity of 1.62E+05 Ci.

In the fuel handling accident radiological consequences analysis both release scenarios are considered, namely inside containment and inside the fuel building. For both release scenarios, Kr-85 contributes less than 0.03% of the total dose at each location (exclusion area boundary, low population zone, control room, and technical support center). The dose increase resulting from an increase of ~27% of Kr-85 activity (gap fraction of 0.30 vs 0.38), would be more than offset by decreasing the activity releases for other nuclides when modeling the ANSI/ANS-5.4-2011 gap fractions.

Therefore, the gap fractions used for WCGS, which are based on RG 1.25, as modified by NUREG/CR-5009, are conservative in comparison to the gap fractions using ANSI/ANS-5.4-2011.

b) The USAR text is revised to state that the use of RG 1.25 is being specified in Table 15B-1 [RG 1.183 Conformance Table].

ARCB-RAI-19

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

Response:

The current single failure of the humidity control system resulted in a reduced iodine removal efficiency. The fuel handing accident analysis, documented in Section 4.3.12 of Enclosure IV of this LAR, modeled a direct release of activity to the environment with no credit for the filtration of the EES. Therefore, by conservatively not crediting the EES filtration, the analysis results documented in Section 4.3.12 of Enclosure IV of this LAR are more limiting than if the previous single failure of the humidity control system had been retained and a reduced iodine removal efficiency was credited.

Consistent with the current control room radiological consequences calculation models, as discussed on page 15A-8 of the USAR, a failure of one of the filtration fans is assumed at the start of emergency mode of operation and a larger unfiltered inflow to the control room is assumed since only half of the makeup flow to the control room can pass through a filter. After a

defined time of 90 minutes, operator action isolates the failed train and reduces the unfiltered inflow to the control room.

ARCB-RAI-20

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

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

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

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

Response:

The current fuel type for WCNOC (17x17 RFA-2) is generically addressed for a DF of 200 at higher rod internal pressures by WCAP-16072-P-A (Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs, ADAMS Accession Number ML042510053). WCAP-16072-P-A was submitted to the NRC for review and approval, and included an evaluation of iodine decontamination factors for fuel rod pressures up to 1500 psig. In the Final Safety Evaluation for WCAP-16072-P-A (ADAMS Accession Number ML041270102), the NRC stated that, "…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 supersede SG 25 for alternative source terms and TID14844 source terms, respectively."

WCAP-16072-P-A was prepared for CE NSSS plant fuel designs. However, the justification for the continued applicability of a DF of 200 was based on evaluations performed in WCAP-7518-L "Radiological Consequences of a Fuel Handling Accident" which did not distinguish between fuel types. WCAP-7518-L provides a method for calculating an iodine DF given a bubble rise time which is dependent on rod internal pressures. This method is not fuel type specific and was used and discussed in WCAP-16072-P-A to justify the continued applicability of a DF value of

200. Therefore, the justification provided in WCAP-16072-P-A is applicable to all Westinghouse fuel types.

It should be noted that Regulatory Guide 1.183 refers to NRC Staff Technical Paper "Evaluation of Fission Product Release and Transport for a Fuel Handling Accident" (ADAMS Accession Number 8402080322). WCAP-7518-L is discussed in the NRC Staff Technical Paper as being submitted to the NRC to present new experimental data pertaining to the consequences of a refueling accident.

Also, the NRC has previously approved the use of a DF of 200 for fuel pressures up to 1500 psig (reference Indian Point Nuclear Unit No. 3 – Issuance of Amendment Re: Full Scope Adoption of Alternative Source Term, ADAMS Accession Number ML050750431). In the NRC approval of this amendment, the NRC noted that the DF for elemental iodine would remain above 400 for fuel pressures at 1500 psig. This is greater than the DF of 285 for elemental iodine used in the analysis to obtain an overall iodine DF of 200. Therefore, the use of a DF of 200 remains conservative and is appropriate for use.

ARCB-RAI-21

Enclosure 6, page 4-69 of the submittal states: "The decay time used in determining the inventory of the damaged rods is 76 hours. Thus, the analysis supports the TS limit [emphasis added] of 76 hours decay time prior to fuel movement." The staff agrees that the decay time appears to meet Criterion 2 of 10 CFR 50.36, "Technical Specifications," but could not locate the TS limit of 76 hours. Please state where in the TSs this limit exists.

Response:

The sentence on page 4-69 of Enclosure VI of Reference 1 is in error. The 76 hour decay time prior to fuel movement limit is not specified in the Technical Specifications. In the conversion to the improved Technical Specifications (Reference 6), discussion of change (DOC) 3-01-R proposed the relocation of specification 3/4.9.3, "Decay Time," to a licensee controlled document. The NRC approved the improved Technical Specifications in License Amendment No. 123 (Reference 7). The safety evaluation for License Amendment No. 123 states, in part:

> 2. CN 3-01-R [current Technical Specifications (CTS)] 3/4.9.3, Reactor Decay Time (CTS 3/4.9)
>
> The requirements in CTS 3/4.9.3 on the decay time that the reactor core must be subcritical before there is movement of irradiated fuel in the reactor core are being relocated to the TRM. This LCO requires the reactor to be subcritical for 100 hours to allow the radioactive decay of the short-lived fission products. The screening criteria for including the requirements in the ITS have been satisfied for Criterion 2 since decay time is consistent with the assumptions used in an accident analysis; however, the activities necessary to be performed at WCGS before commencing movement of irradiated fuel

ensure that 100 hours of subcriticality will elapse before there is movement of irradiated fuel in the core. Therefore, because the CTS is not required to assure that 100 hours have elapsed prior to fuel movement, the decay time LCO and SRs in the CTS may be relocated to the TRM, a licensee-controlled document outside TS. The TRM is included by reference in the USAR and is an acceptable licensee-controlled document; therefore, the relocation is acceptable.

The two relocated specifications from the CTS discussed above are not required to be in the ITS because they do not fall within the criteria for mandatory inclusion in the TS in 10 CFR 50.36(c)(2)(ii). They are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. In addition, the NRC staff finds that sufficient regulatory controls exist under 10 CFR 50.59 to maintain the effect of the provisions in these specifications. The NRC staff has concluded that appropriate controls have been established for all of the current specifications that are being moved to the USAR.

The decay time requirement was relocated to USAR Section 9.1.4.2.3. Additionally, the TS 3.9.7 Bases, "Refueling Pool Water Level," discussed the minimum decay time prior to fuel handling.

ARCB-RAI-22:

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

Response:

The assumptions and inputs for the fuel handling accident within containment and outside of containment are identical with the exception of control room and technical support center atmospheric dispersion factors associated with the different release points.

ARCB-RAI-23:

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

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

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

Response:

a) Based on the NRC question, WCNOC re-reviewed the release point and receptor combinations and determined more limiting pairs should be used. Two additional release points were determined to need atmospheric dispersion factor calculations: (1) release from a reactor building wall diffuse area source and (2) release from the radwaste building. In addition, it was also discovered during the review (of release point and receptor combinations) that atmospheric dispersion factors needed to be determined for one additional receptor: the control room normal HVAC intake.

   Due to the new atmospheric dispersion factor calculations, four of the presented accident and transport path combinations were determined to not be using the most limiting combination of release point and receptor. The affected accident and transport path combinations are: (1) rod ejection accident containment leakage transport path to the control room emergency intake vent, (2) loss of coolant accident containment leakage transport path to the control room emergency intake vent, (3) waste gas decay tank failure transport path to the control room emergency intake vent and TSC intake vent, and (4) liquid waste tank failure transport path to the control room emergency intake vent and TSC intake vent. These accident and transport path combinations have been updated and are included in the response to item "c)" below, and also contained in Enclosure IV of this LAR, Section 4.1.2.4 and Table 4.1.2-5.

b) The updated (refer to response "a)" above) Table 4.1.2-3 and updated individual accident tables (Table 4.3-6 through 4.3-15) have been included in Enclosure IV of this LAR, and have been confirmed to contain the same information. The updated version of the Table 4.1.2-3, which has been renumbered to 4.1.2-3(a), is included below for reference.

Table 4.1.2-3(a) Calculated χ/Q (sec/m^3) for the Emergency Control Room Intake Vent

Equipment Hatch to Emergency Control Room Air Intake	
0 to 2 hours	5.44E-04
2 to 8 hours	4.35E-04
8 to 24 hours	1.62E-04
1 to 4 days	1.22E-04
4 to 30 days	8.70E-05
Unit Vent Stack to Emergency Control Room Air Intake	
0 to 2 hours	6.12E-04
2 to 8 hours	4.38E-04
8 to 24 hours	1.79E-04
1 to 4 days	1.14E-04
4 to 30 days	8.94E-05
MSSVs/ARVs to Emergency Control Room Air Intake	
0 to 2 hours	1.04E-03
2 to 8 hours	7.46E-04
8 to 24 hours	3.03E-04
1 to 4 days	1.90E-04
4 to 30 days	1.39E-04
RWST Vent to Emergency Control Room Air Intake	
0 to 2 hours	6.80E-04
2 to 8 hours	6.19E-04
8 to 24 hours	2.27E-04
1 to 4 days	1.96E-04
4 to 30 days	1.53E-04
TDAFW Exhaust to Emergency Control Room Air Intake	
0 to 2 hours	5.17E-04
2 to 8 hours	3.99E-04
8 to 24 hours	1.60E-04
1 to 4 days	1.00E-04
4 to 30 days	7.21E-05
Radwaste Building to Emergency Control Room Air Intake	
0 to 2 hours	6.92E-04
2 to 8 hours	6.06E-04
8 to 24 hours	2.39E-04
1 to 4 days	2.00E-04
4 to 30 days	1.47E-04
Reactor Building Wall to Emergency Control Room Air Intake	
0 to 2 hours	6.02E-04
2 to 8 hours	4.35E-04
8 to 24 hours	1.81E-04
1 to 4 days	1.29E-04
4 to 30 days	9.65E-05

c) Enclosure IV of this LAR, Table 4.1.2-5, included for reference below, provides the release points considered in the calculation of the χ/Q values for each transport path of each accident. The table is structured according to the accident type and associated transport paths. The accident types include (1) main steam line break (two transport paths), (2) loss of non-emergency AC power (one transport path), (3) locked rotor (one transport path), (4) rod ejection (two transport paths), (5) letdown line break (one transport path), (6) steam generator tube rupture (one transport path), (7) loss of coolant accident (four transport paths), (8) waste gas decay tank failure (one transport path), (9) liquid waste tank failure (one transport path), and (10) fuel handling accident (two transport paths). Some accidents have multiple transport paths. The χ/Q values assigned to a given accident depend on the transport paths associated with that accident. Accidents with the same transport path are assigned the same χ/Q values.

As explained in Enclosure IV of this LAR, Section 4.1.2.4, for each accident, transport paths and release points are identified. Transport paths refer to paths such as primary-to-secondary leakage, primary-to-secondary break, containment leakage, ECCS leakage, RWST back-leakage, containment purge, direct release to auxiliary building, and direct release to atmosphere. The release points refer to the release points whose χ/Q values for CR and TSC intakes have been determined. These release points include the reactor building equipment hatch, reactor building wall diffuse source, unit vent stack, TDAFW exhaust vent, MSSV/ARV vents, RWST vent, and radwaste building. If a transport path has a single release point, the χ/Q associated with that release point is the limiting value. Some transport paths may have multiple release points. For such cases, the limiting χ/Q values are assumed to be the maximum of all release points associated with that transport path and release period. A site diagram with the considered release points and receptor points labeled is shown in Figure 4.1.2-1 below, and contained in Enclosure IV of this LAR, Section 4.1.2.

The containment leakage path considers three release points (equipment hatch, unit vent stack, and diffuse reactor building wall). For the containment leakage path, the limiting χ/Q values applied to the emergency control room intake are from the unit vent stack for a period from 0 to 8 hours, and from the reactor building wall for a period from 8 hours to 30 days. The limiting χ/Q values for the normal control room intake are from the unit vent stack for all periods. The limiting χ/Q values for the technical support intake are from the equipment hatch for all periods.

The primary-to-secondary leakage to the intact SGs is another example of a transport path with multiple release points. The path through the intact SGs leads to the release points at the MSSV/ARV vents and the TDAFW exhaust vent. For this case, the limiting χ/Q values are assigned to the larger values of the two release points.

For the main steam line break accident, the primary-to-secondary leakage to the intact SGs and the faulted SG are the two transport paths that lead to different release points to the environment. The path through the intact SGs leads to the release points at the MSSV/ARV vents and the TDAFW exhaust vent. The limiting χ/Q values for the intact SG release path are assumed the larger values of the two release points. However, the path through the faulted SG releases radionuclides into the auxiliary building if a steam line break occurs outside containment. The main steam line break is assumed to occur outside of containment because a break outside of the containment will bound any break inside containment. The radionuclides released from the break are drawn by the building ventilation system to the unit vent stack and eventually exhausted to the environment through the unit vent stack. Hence, the χ/Q values for the broken steam line release path are the values for the unit vent.

For the loss of non-emergency AC power accident, the primary-to-secondary leakage to intact SGs transport path leads to two release points at the MSSV/ARV vents and the TDAFW exhaust vent. The χ/Q values for this release path are assumed as the larger value of the two release points.

For the locked rotor accident, the primary-to-secondary leakage to SGs transport path leads to two release points at the MSSV/ARV vents and the TDAFW exhaust vent. The χ/Q values for this release path are assumed as the larger value of the two release points.

For the rod ejection accident there are two transport paths, containment leakage to the environment, and primary to secondary leakage. The χ/Q values for the containment leakage release path are assumed as the maximum value of the three release points: (1) equipment hatch, (2) reactor building wall, and (3) unit vent stack. The χ/Q values for the primary to secondary leakage release path are assumed as the larger value of the MSSV/ARV vents and the TDAFW exhaust vent release points.

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 values for this case are those of the unit vent stack.

For the steam generator tube rupture accident, the primary-to-secondary break to the ruptured SG is the major transport path to the environment. The χ/Q values for the ruptured SG release path are assumed the larger value of the MSSV/ARV vents and the TDAFW exhaust vent.

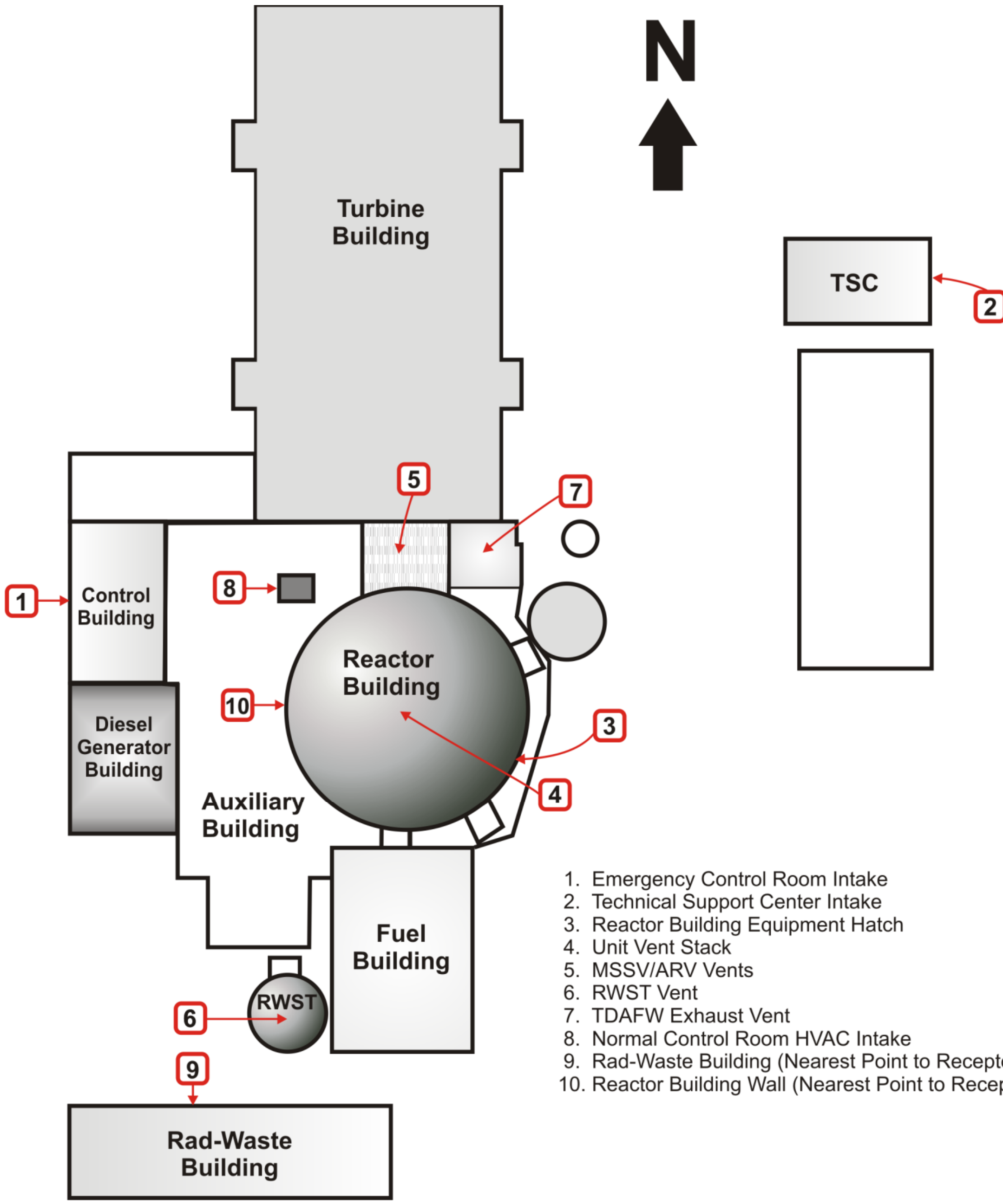
For the loss of coolant accident, there are four transport paths. The first path is the containment leakage transport path to the environment. The χ/Q values for this transport path are assumed as the maximum value of the three release points: (1) equipment hatch, (2) reactor building wall, and (3) unit vent stack. The second transport path is the ECCS leakage path. The χ/Q values for this path are those of the unit vent stack. The third transport path is the RWST back-leakage path. The χ/Q values for this path are of the RWST vent. The fourth transport path is the containment purge path. The containment purge system vents the containment atmosphere

through the containment exhaust penetrations up to the unit vent stack. Therefore, the χ/Q values for this path are those of the unit vent stack.

For the waste gas decay tank failure, direct release from the radwaste building is the only credited transport path to the environment. The χ/Q values for this case are those of the radwaste building.

For the liquid waste tank failure, direct release from the radwaste building is the only credited transport path to the environment. The χ/Q values for this case are those of the radwaste building.

For the fuel handling accident, two possible accident types are considered: (1) in containment and (2) in fuel building. For a fuel handling accident occurring in containment, the transport path is through the open equipment hatch. For a fuel handling accident occurring in the fuel building, the radionuclides released are drawn by the building ventilation system to the unit vent stack and eventually exhausted to the environment through the unit vent stack.



1. Emergency Control Room Intake
2. Technical Support Center Intake
3. Reactor Building Equipment Hatch
4. Unit Vent Stack
5. MSSV/ARV Vents
6. RWST Vent
7. TDAFW Exhaust Vent
8. Normal Control Room HVAC Intake
9. Rad-Waste Building (Nearest Point to Receptor)
10. Reactor Building Wall (Nearest Point to Receptor)

Figure 4.1.2-1. Diagram of Source and Receptor Locations for Wolf Creek

Table 4.1.2-5 Accident Release Sources

Accident	Transport Path	Release Points	Limiting χ/Q
Main Steamline Break	Primary to Secondary Leakage to Intact SG	MSSVs/ARVs TDAFW Exhaust	Maximum of all Associated Release Points
	Primary to Secondary Leakage to Faulted SG for a Break Outside of Containment	Unit Vent	Unit Vent
	Primary to Secondary Leakage to Faulted SG for a Break Inside of Containment (This transport path is not used because the break outside of containment is bounding)	Equipment Hatch Unit Vent Reactor Building Wall (Diffuse Source)	Maximum of all Associated Release Points
Loss of Non-Emergency AC Power	Primary to Secondary Leakage	MSSVs/ARVs TDAFW Exhaust	Maximum of all Associated Release Points
Locked Rotor	Primary to Secondary Leakage	MSSVs/ARVs TDAFW Exhaust	Maximum of all Associated Release Points
Rod Ejection	Containment Leakage	Equipment Hatch Unit Vent Reactor Building Wall (Diffuse Source)	Maximum of all Associated Release Points
	Primary to Secondary Leakage	MSSVs/ARVs TDAFW Exhaust	Maximum of all Associated Release Points
Letdown Line Break	Release Outside of Containment	Unit Vent	Unit Vent
Steam Generator Tube Rupture	Primary to Secondary Break	MSSVs/ARVs TDAFW Exhaust	Maximum of all Associated Release Points

Table 4.1.2-5 Accident Release Sources (cont.)

Accident	Transport Path	Release Points	Limiting χ/Q
Loss of Coolant Accident	Containment Leakage	Equipment Hatch	Maximum of all Associated Release Points
		Unit Vent	
		Reactor Building Wall (Diffuse Source)	
	ECCS Leakage	Unit Vent	Unit Vent
	RWST Back-Leakage	RWST Vent	RWST Vent
	Containment Purge	Unit Vent	Unit Vent
Waste Gas Decay Tank Failure	Release to Atmosphere	Radwaste Building	Radwaste Building
Liquid Waste Tank Failure	Release to Atmosphere	Radwaste Building	Radwaste Building
Fuel Handling Accident	Release In Containment	Equipment Hatch	Equipment Hatch
	Release In Fuel Building	Unit Vent	Unit Vent

ARCB-RAI-24

In a letter dated November 7, 2013 (ADAMS Accession No. ML 13246A358), the NRC informed the Technical Specifications Task Force of concerns that the NRC staff had recently identified during a review of plant-specific license amendments requesting adoption of three travelers including traveler TSTF-51, Revision 2, "Revise Containment Requirements during Handling Irradiated Fuel and Core Alterations."

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

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

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

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

Appendix B of RG 1.183, Regulatory Position 1.1 states:

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

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

Response:

Reference 1 included the adoption of TSTF-51-A, Revision 2. WCNOC became aware of the NRC concerns with TSTF-51 in early October 2013 after Reference 1 had been submitted to the NRC. With the subsequent issuance of a letter dated November 7, 2013 (Reference 8), WCNOC communicated with the NRC Project Manager on November 14, 2013 by electronic mail that consideration would be given to withdrawing the changes proposed by TSTF-51-A depending on the significance of the impact to Reference 1. WCNOC withdrew the license amendment request (Reference 1) by WCNOC letter dated June 18, 2014 (Reference 3). The resubmittal of the license amendment request excludes the adoption to TSTF-51, Revision 2.

ARCB-RAI-25

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

Response:

Regulatory Position 5.5.4 in Appendix E of Regulatory Guide 1.183, which specifies the iodine partition of 100 in the steam generators, does not distinguish between elemental and organic iodine. As a result, the partition factor of 100 was applied to both forms of iodine being released from the steam generators for all applicable analyses. The NRC approved this modeling of partitioning on iodine releases from the steam generators in previous AST submittals (reference Point Beach Nuclear Plant (PBNP), Units 1 and 2 -Issuance of License Amendments Regarding Use of Alternate Source Term (TAC Nos. ME0219 and ME0220), ADAMS Accession Number ML110240054).

ARCB-RAI-26

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

Response:

The modeling of progeny has been included in the updated results presented in Enclosure IV of this LAR, Section 4.3. The comments in Enclosure IV of this LAR, Table A for Regulatory Position 4.1.1 have been updated to the following response:

> “The dose calculations determine the TEDE and consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences.”

ARCB-RAI-27

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

Response:

A spike in cesium associated with the iodine spikes (both pre-accident and accident-initiated) in the WCGS radiological consequences analyses (main steamline break, steam generator tube rupture, loss of AC power, and letdown line break) was not explicitly modeled, consistent with methods used in previously approved AST submittals (e.g., Point Beach Units 1 and 2 AST Approval, April 14, 2011 under ADAMS Accession Number ML110240054). However the initial cesium activity modeled in the RCS bounds the cesium which would be released to the RCS in association with iodine spiking.

The cesium activity initially modeled in the RCS for the above accident analyses was that associated with the design basis fuel defect level of 1%. The initial RCS iodine activity was scaled down to the Technical Specification 3.4.16, “RCS Specific Activity,” limit of ≤ 1.0 μCi/gm dose equivalent (DE) I-131 as identified in Enclosure IV of this LAR, Section 4.3 and explained further in the response to ARCB-RAI 37. As shown in the response to ARCB-RAI 37, the total isotopic dose equivalence for the iodines considered is 4.22, which means that the 1% fuel defect iodine activity is reduced by a factor of 4.22 to be at the Technical Specification 3.4.16 limit of ≤ 1.0 μCi/gm DE I-131. Therefore, the fuel defect level corresponding to the Technical Specification limit is approximately 0.24% fuel defects.

Tables 1 and 2 convert the fuel defect level activities to corresponding nuclide masses. The RCS mass used in the calculation is the mass used in the dose calculations (1.81E+08 gm). Note that the calculations were performed in a spreadsheet and additional decimal places were not included in the tables for ease of presentation.

A sample calculation for Cs-134 is as follows:

RCS Activity (Ci) = 1% Fuel Defect Activity * RCS Mass * 1 Ci / 1E6 μCi
= 4.82E+00 μCi/gm * 1.81E+08 gm * 1 Ci / 1E6 μCi = 8.72E+02 Ci

RCS Activity (Bq) = RCS Activity (Ci) * 3.7E+10 Bq/Ci
= 8.72E+02 Ci * 3.7E+10 Bq/Ci = 3.23E+13 Bq

Number of atoms = RCS Activity (Bq) * Half-life (sec) / ln(2)
= 3.23E+13 Bq * 6.50E+07 sec / ln(2) = 3.03E+21 atoms

Mass (gm) = Number of atoms * Atomic Mass Number (gm/mole) / 6.022E+23 (atoms/mole)
= 3.03E+21 atoms * 134 gm/mole / 6.022E+23 atoms/mole = 6.74E-01 gm

Table 1 Cesium Mass at 1% Fuel Defect Level

Nuclide	1% Fuel Defect Activity (µCi/gm)	RCS Activity (Ci)	RCS Activity (Bq)	Half-life (sec)	Number of atoms	Mass (gm)
Cs-134	4.82E+00	8.72E+02	3.23E+13	6.50E+07	3.03E+21	6.74E-01
Cs-136	4.35E+00	7.87E+02	2.91E+13	1.13E+06	4.75E+19	1.07E-02
Cs-137	2.68E+00	4.85E+02	1.79E+13	9.46E+08	2.45E+22	5.57E+00
Cs-138	1.16E+00	2.10E+02	7.77E+12	1.93E+03	2.16E+16	4.96E-06
					Total:	**6.26E+00**

Table 2 Cesium Mass at Technical Specification Limit Fuel Defect Level

Nuclide	0.24% Fuel Defect Activity (µCi/gm)	RCS Activity (Ci)	RCS Activity (Bq)	Half-life (sec)	Number of atoms	Mass (gm)
Cs-134	1.16E+00	2.09E+02	7.75E+12	6.50E+07	7.26E+20	1.62E-01
Cs-136	1.04E+00	1.89E+02	6.99E+12	1.13E+06	1.14E+19	2.57E-03
Cs-137	6.43E-01	1.16E+02	4.31E+12	9.46E+08	5.88E+21	1.34E+00
Cs-138	2.78E-01	5.04E+01	1.86E+12	1.93E+03	5.19E+15	1.19E-06
					Total:	**1.50E+00**

As shown, the amount of excess cesium mass in modeling 1% fuel defects as opposed to the fuel defects of 0.24% is 4.76 gm. The amount of excess was determined for the radiocesium nuclides considered in the dose analyses (Cs-134, Cs-136, Cs-137, and Cs-138).

For the pre-accident spike scenario, the total iodine mass modeled in the RCS at the spike limit of 60 µCi/gm DE I-131 is calculated. The mass contributions of both I-127 and I-129, as well as the mass contributions of iodine modeled in the dose analyses (I-130 through I-135), are calculated. Since I-127 is stable, its contribution to 1% fuel defects is given in units of grams of I-127 per gram of RCS coolant.

Table 3 converts the fuel defect level activities to corresponding nuclide masses. The pre-accident spike activity level used is 60 µCi/gm DE I-131 and the RCS mass used in the calculation is the mass used in the dose calculations (1.81E8 gm). The calculations were performed in a spreadsheet and additional decimal places were not included in the tables for ease of presentation.

Table 3 Iodine Mass in RCS for Pre-Accident Iodine Spike

Nuclide	Activity at 1.0% Fuel Defect Level (µCi/gm)	Isotopic Activity Scaled to 1.0 µCi/gm DE I-131 (µCi/gm)	Isotopic Activity at 1.0 µCi/gm DE I-131 (Ci)	Isotopic Activity at 60 µCi/gm DE I-131 (Ci)	Isotopic Activity at 60 µCi/gm DE I-131 (Bq)	Half-life (sec)	Number of atoms	Mass (gm)
I-127	1.24E-10 (gm/gm)	2.94E-11 (gm/gm)	5.32E-03 (gm)	3.19E-01 (gm)	N/A	N/A	N/A	3.19E-01
I-129	7.17E-08	1.70E-08	3.07E-06	1.84E-04	6.82E+06	4.95E+14	4.87E+21	1.04E+00
I-130	4.65E-02	1.10E-02	1.99E+00	1.20E+02	4.43E+12	4.45E+04	2.84E+17	6.13E-05
I-131	3.28E+00	7.77E-01	1.41E+02	8.44E+03	3.12E+14	6.95E+05	3.13E+20	6.81E-02
I-132	3.39E+00	8.03E-01	1.45E+02	8.72E+03	3.23E+14	8.28E+03	3.85E+18	8.45E-04
I-133	5.04E+00	1.19E+00	2.16E+02	1.30E+04	4.80E+14	7.49E+04	5.18E+19	1.14E-02
I-134	7.30E-01	1.73E-01	3.13E+01	1.88E+03	6.95E+13	3.16E+03	3.17E+17	7.05E-05
I-135	2.85E+00	6.75E-01	1.22E+02	7.33E+03	2.71E+14	2.38E+04	9.31E+18	2.09E-03
							Total	**1.45E+00**

Since the atomic masses of cesium and iodine are similar and 95% of the iodine is in particulate form (i.e., CsI), then the total cesium which would be modeled by a pre-accident iodine spike is 1.37 gm (1.45 gm of iodine * 0.95 fraction of CsI to total iodine), which is less than the excess radiocesium initially modeled in the RCS.

For the accident-initiated spike scenario, the total iodine mass in the gap of the defective fuel rods is calculated. The mass contributions of both I-127 and I-129, as well as the mass contributions of iodine modeled in the dose analyses (I-130 through I-135), are calculated. Since I-127 is stable, its contribution in the core is given in units of grams of I-127.

Table 4 converts the iodine activity available in the core to corresponding nuclide masses that are available for release from the fuel gap in the defective fuel rods. The fuel gap fractions of 12% for I-131 and 10% for the remaining iodines are used and are conservative gap fractions based on those used in the fuel handling accident analysis. Also, the fuel defect level of 0.24% (described above) is used. Note that the calculations were performed in a spreadsheet and additional decimal places were not included in the tables for ease of presentation.

Table 4 Iodine Mass Available for Release from Fuel Gap in Accident-Initiated Iodine Spike

Nuclide	Activity Available In Core (Ci)	Gap Fraction	Activity Available In Gap (Ci)	Activity Available in Gap (Bq)	Fuel Defect level (%)	Activity Available in Defective Fuel (Bq)	Half Life (sec)	Number of atoms	Mass (gm)
I-127	4.31E+03 (gm)	0.1	4.31E+02 (gm)	N/A	0.24	1.03E+00 (gm)	N/A	N/A	1.03E+00
I-129	2.57E+00	0.1	2.57E-01	9.51E+09	0.24	2.28E+07	4.95E+14	1.63E+22	3.49E+00
I-130	1.98E+06	0.1	1.98E+05	7.33E+15	0.24	1.76E+13	4.45E+04	1.13E+18	2.44E-04
I-131	1.01E+08	0.12	1.21E+07	4.48E+17	0.24	1.08E+15	6.95E+05	1.08E+21	2.35E-01
I-132	1.49E+08	0.1	1.49E+07	5.51E+17	0.24	1.32E+15	8.28E+03	1.58E+19	3.46E-03
I-133	2.10E+08	0.1	2.10E+07	7.77E+17	0.24	1.86E+15	7.49E+04	2.02E+20	4.45E-02
I-134	2.36E+08	0.1	2.36E+07	8.73E+17	0.24	2.10E+15	3.16E+03	9.55E+18	2.13E-03
I-135	2.00E+08	0.1	2.00E+07	7.40E+17	0.24	1.78E+15	2.38E+04	6.10E+19	1.37E-02
								Total:	**4.82E+00**

Since the atomic masses of cesium and iodine are similar and 95% of the iodine is in particulate form (i.e., CsI), then the total cesium which would be modeled by an accident-initiated iodine spike is 4.58 gm (4.82 gm of iodine * 0.95 fraction of CsI to total iodine), which is less than the excess radiocesium modeled in the RCS.

Therefore, the dose impact of modeling the design fuel defect level cesium activity in the RCS instead of scaling to the fuel defect level activity bounds the impact of modeling an increase in radiocesium resulting from iodine spiking.

It should be noted that the core composition of cesium isotopes includes approximately 40% Cs-133 which is a stable nuclide and has no dose impact. Assuming that the core composition of cesium is similar to that released from the fuel gap to the RCS as CsI, then the total radiocesium which would be released during an iodine spike is approximately 60% of that in the calculations above. Therefore, the calculations above are conservative.

ARCB-RAI-28

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

Response:

The transport model described in Regulatory Positions 5.5 and 5.6 in Appendix E of Regulatory Guide 1.183 were used for the Locked Rotor and Rod Ejection analyses with the following exceptions:

- Position 5.5.1: For the Rod Ejection and Locked Rotor radiological consequences analyses, steam generator dryout does not occur due to the secondary system remaining intact so no consideration for flashing of primary-to-secondary leakage was modeled.

- Position 5.5.2: As there is no flashing of primary-to-secondary leakage, no credit was taken in the Rod Ejection and Locked Rotor radiological consequences analyses for scrubbing of flashed leakage in the steam generators.

- Position 5.6: Comments provided in Table D of Enclosure VI (now Enclosure IV of this LAR) for Regulatory Position 5.6 state that the issue for tube uncovery was addressed in WCAP-13247 [The Topical Report on the Methodology for the Resolution of the Steam Generator Uncovery Issue is included as part of the response to ARCB-RAI-36 for non-SGTR events (Rod Ejection and Locked Rotor included).] It was concluded in WCAP-13247 and confirmed in the NRC response to WCAP-13247 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 as

noted in Table D. Therefore, the Rod Ejection and Locked Rotor radiological consequences analyses do not model tube uncovery.

ARCB-RAI-29:

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

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

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

Response:

This assumption was not credited in the previous WCGS analyses of record. This modeling is justified by the WCGS Emergency Management Guidelines (EMGs), which instruct the operators to maintain the level in the steam generators between 6% and 50% of narrow range span (NRS). The minimum mass modeled in the analysis is associated with a NRS SG water level of 0% corresponding to hot zero power conditions. This is conservative given the EMGs instruct the operators to maintain level of at least 6% of NRS.

The following wording is added to Enclosure IV, Sections 4.3.4.2.2, 4.3.5.2.2, and 4.3.6.2.2.2:

"…as justified by the Emergency Management Guidelines (EMGs). The mass used in the analysis after 2 hours was calculated at 0% narrow range to appropriately bound the just on span level of 6% narrow range."

ARCB-RAI-30

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

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

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

Response:

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

- LOCA
- MSLB
- SGTR
- Rod Ejection
- Locked Rotor
- Fuel Handing Accident

For the SGTR analysis, the limiting single failure is a failed open steam generator ARV on the ruptured steam generator, which is modeled in the mass releases calculated in Section 2.7.2 of WCAP-17658-NP, Revision 1 (Enclosure I of this LAR). The single failure of a failed open steam generator ARV is conservatively modeled in addition to the failure of one of the CREVS filtration fans.

For the remaining dose analyses, no CREVS actuation is generated due to the event and thus the failure of one of the CREVS filtration fans is not modeled. These accidents include:

- Tank ruptures
- Letdown line break
- Loss of AC power

These accidents model a release directly to the environment and do not credit filtration from either the EES or from the CREVS. Thus, there is not a single failure that would decrease the effectiveness of the exhaust or intake filtration and therefore an explicit single failure is not modeled. This is consistent with the current licensing basis AOR.

ARCB-RAI-31:

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

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

a) *If previously the tank was assumed to accumulate activity over 40 years, please explain why the activity from one fuel cycle is conservative and consistent with the RG 1.24 above regulatory position of using the maximum content of the decay tank. Please justify the proposed change in assumptions regarding the tank contents and state why they are conservative.*

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

Response:

a) The methods and assumptions used to determine the proposed gas decay tank source terms are consistent with, and somewhat more conservative than Regulatory Guide 1.24 as discussed below.

As stated in Regulatory Guide 1.24 Section C.1, Regulatory Position, the assumptions related to the release of radioactive gases from the postulated failure of a gaseous waste storage tank should be,

a. “The reactor has been operating at full power with one percent defective fuel and a shutdown to cold condition has been conducted near the end of an equilibrium core cycle. As soon as possible after shutdown, all noble gases have been removed from the primary cooling system and transferred to the gas decay tank that is assumed to fail.

b. The maximum content of the decay tank assumed to fail should be used for the purpose of computing the noble gas inventory in the tank. Radiological decay may be taken into account in the computation only for the minimum time period required to transfer the gases from the primary system to the decay tank.

c. The failure is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank to the building. The assumption of the release of the noble gas inventory from only a single tank is based on the premise that all gas decay tanks will be isolated from each other whenever they are in use.

d. All of the noble gases are assumed to leak out of the building at ground level over a two hour time period."

The WCGS design contains 6 decay tanks that are available for containing waste gases during plant operation and, as noted in the Discussion Section B of Regulatory Guide 1.24,

> "… The radioactive components are principally the noble gases krypton and xenon, the particulate daughters of some of the krypton and xenon isotopes, and trace quantities of the halogens. With the exception of krypton-85, the longest half-life of the principal noble gas radionuclides present in reactor effluents is 5.27 days (xenon-133). Thus, storage of these gases for a period of 60 days will essentially eliminate by decay all of the radionuclides except krypton-85".

Thus, the krypton-85 activity that is generated during plant operations prior to cold shutdown will have been discharged from the tank(s) following the storage period during which the short lived species are allowed to decay. If the krypton activity is not discharged, it will be distributed among one or all of the 6 "normal operation" decay tanks.

The design also includes 2 "shutdown" gas decay tanks. The gaseous inventory in the primary coolant system is transferred to one (or both) of these tanks via the volume control tank (VCT) "as soon as possible after shutdown." Since the shutdown tank will contain a relatively large amount of (primarily) short-lived activity, this is the tank that is assumed to fail. Also, for additional conservatism, all of the krypton-85 activity generated during the previous core cycle (from both normal operation and shutdown operations) is assumed to be present in the failed tank prior to the release.

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.

b) All of the source terms are based on the conservative assumption that the plant has operated for a complete equilibrium core cycle with fuel cladding defects in the fuel rods that generate one percent of the core thermal power. A power uncertainty (i.e., calorimetric power uncertainty) factor of 1.02 is considered in the calculation of the core activities; an assumption that meets the requirements of Regulatory Guide 1.195 and Regulatory Guide 1.183, i.e. "The inventory of fission products … should be based on … an assumed core power equal to the current licensed rated thermal power times the emergency core cooling system (ECCS) evaluation uncertainty." In addition, all calculated activity values are increased be a factor of [ ]^{a,c} in order to account for potential core design parameter differences in future core designs. This factor is generally referred to as a "fuel management multiplier" and is intended to provide "bounding" values relative to future core designs as applied to accident analyses. Further, the primary coolant concentrations are conservatively assumed to be at the peak values calculated over the operating period for the case with no purge of the volume control tank (VCT) during power operation.

The waste gas decay tank sources are based on solving a set of equations that model the shutdown degassing of the primary coolant in which the isotopic inventory of the gas decay tank varies with time. The letdown of the primary coolant is assumed to occur at the maximum letdown flow rate with coolant degassing operations accomplished by "burping" of the volume control tank vapor space at 3-hour intervals. The maximum activities over the degassing period are conservatively selected in defining the limiting gas decay tank inventory. 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. The Kr-85 activity inventory is treated differently in that the total activity released to the coolant during the fuel cycle is assumed to be transferred to the gas decay tank.

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. The recycle holdup tank activity release is used in analyzing the whole body dose impacts of the accident, and the release of activity from the hypothetical waste tank is considered in determining the thyroid dose consequences of a postulated accident.

The source terms for the tanks consider primary coolant activities as an input and apply appropriate reduction factors in order to determine the tank activity inventories. The reduction factors are based on decay and the reduction in inventory due to removal by the demineralizers which are located upstream of the tank. The contribution of the parent nuclide is also considered.

Two sources of activity are considered in the evaluation of the recycle holdup tank sources; i.e., the Reactor Coolant System (RCS) and the Reactor Coolant Drain Tank (RCDT). The liquid goes through two mixed bed demineralizers between the exit from the RCS and the recycle holdup tank. The intermittent flow through the cation demineralizer located between the two mixed bed demineralizers is conservatively ignored. Also, there is one demineralizer between

the outlet of the RCDT and the recycle holdup tank. Thus, the associated Decontamination Factors (DFs) considered in the analysis are:

For the noble gases, no reduction by the demineralizers is considered, i.e., the DF = 1

For Br and Iodine, a factor of 10 reduction across each demineralizer is considered, i.e.,

For the RCS to recycle holdup tank: the DF = 10 x 10 = 100
For the RCDT: the DF = 10

The concept of a “hypothetical liquid waste tank” was originally developed by WCNOC in order to replace sources associated with the evaporator bottoms tank, since the WCGS does not use the evaporators and the evaporator bottoms tank was no longer considered to be the bounding case for the thyroid dose analysis. Although not associated with an existing piece of equipment, the continued use of sources associated with a “hypothetical liquid waste tank” is considered to be appropriate. The “hypothetical waste tank” receives fluid from three separate tanks:

- A recycle holdup tank (RHUT),
- A waste holdup tank (WHUT), and
- A floor drain tank (FDT).

It is conservatively assumed that the input to each of the tanks is based on continuous full power operation with equilibrium iodine activity levels in the primary coolant. Based on this activity concentration, it is assumed that each of the feed tanks is filled and drained such that a constant volume in the feed tank is maintained and the activity concentration reaches an equilibrium level. Thus, the feed rate into each tank equals the discharge rate from the tank into the hypothetical tank and the (equilibrium) activity concentration in the feed tank comprises the input of activity into the hypothetical waste tank. Finally, equilibrium activity levels are assumed to exist in the hypothetical waste tank based on decay alone.

ARCB-RAI-32

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

Response:

The time to terminate steam generator tube leakage to the intact SGs (i.e., SGs neither faulted nor ruptured) was set to the time at which the steam releases were terminated (12 hours, which is the time that the residual heat removal systems match decay heat and eliminate the need for steaming from the secondary system for all analyses involving steam releases). When the

steam releases are terminated, the activity leakage into the secondary system no longer has an impact on the resulting doses because the activity released to the environment is terminated.

ARCB-RAI-33

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

Response:

The gas-phase mass-transfer coefficient (k_g term) from BNL Technical Report A-3788, dated August 1986 (included in Enclosure VIII of this LAR), was previously used in the Indian Point Unit 2 License Amendment Request (LAR) in the transition to the use of the AST. The Indian Point Unit 2 LAR (reference Proposed Amendment Consisting of Changes to Technical Specification for Containment Air Filtration, Control Room Air Filtration, and Refueling Conditions, ADAMS Accession Number ML993400414) references the Indian Point Unit 2 Licensing Report submitted to the NRC in the October 8, 1999 letter (reference Radiological Consequences of Accidents for the Indian Point Nuclear Generating Station Unit No. 2 Using Source Term Methodology from NUREG-1465, ADAMS Accession Number ML100430988) which gives a k_g value of 9.84 ft/min (which is consistent with the value used for the WCGS analysis). It was determined by the NRC staff in the Safety Evaluation (reference Indian Point Nuclear Generating Unit No. 2 - Re: Issuance of Amendment Affecting Containment Air Filtration, Control Room Air Filtration, and Containment Integrity During Fuel Handling Operations (TAC No. MA6955, ADAMS Accession Number ML003727500)) that "…the licensee used plant-specific parameters and a very conservative value for the gas phase mass transfer coefficient."

Therefore, the k_g value from BNL Technical Report A-3788 is also appropriate for use in the WCGS analysis.

The BNL report was used as a reference for the k_g value in the proposed revision to SRP Section 6.5.2, Revision 2 (dated April 1987) and was included by the NRC as part of this proposed revision.

From SRP Section 6.5.2, Revision 4: "The first-order removal coefficient by spray, λ_s, may be taken to be λ_S=6K_gTF/VD where K_g is the gas-phase mass-transfer coefficient and T is the time of fall of the drops, which may be estimated by the ratio of the average fall height to the terminal velocity of the mass-mean drop."

Earlier revisions of SRP Section 6.5.2 (Revisions 2 and 3) cite Reference 20 at the end of the above sentence. This statement indicates the Reference 20 document is only meant for the terminal velocity calculations that result in the fall time of the drops. Reference 20 is not cited because it does not contain a recommended k_g value.

ARCB-RAI-34

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

Response:

In the response to ARCB-RAI-24, WCNOC identified that the changes proposed in TSTF-51-A, Rev. 2, are being withdrawn. As such, the changes to TS 3.9.4 are being withdrawn and the Limiting Condition for Operation (LCO) 3.9.4 requirements remain applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

The current LCO 3.9.4 allows the equipment hatch to be open and capable of being closed, one personnel air lock capable of being closed, and penetration flow paths that have direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. These allowances were approved in Amendment No. 146 (Reference 10), Amendment No. 95 (Reference 11), and Amendment No. 135 (Reference 12), respectively. Consistent with the current analysis of record, the radiological consequences AST analysis for a fuel handling accident inside containment did not credit isolation of containment. Administrative controls (previously approved by the NRC) provide reasonable assurance that containment closure as a defense-in-depth measure can be reestablished quickly to limit releases much lower than assumed in the dose calculations.

ARCB-RAI-35

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

Response:

During the NRC review of the Final Safety Analysis Report (FSAR) for issuance of the operating license, Section 15.4.4, "Steam Generator Tube Rupture," of NUREG-0881, Supplement 5, (Reference 13) "Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station, Unit No. 1," specified that the staff will condition the operating license to require satisfactory resolution of this issue before startup following the first refueling outage. This

became License Condition 2.C.(11), Steam Generator Tube Rupture (Section 15.4.4, SSER #5) which required Kansas Gas and Electric to submit for NRC review and approval an analysis that demonstrated that the steam generator tube rupture (SGTR) analysis presented in the FSAR is the most severe case with respect to release of fission products and calculated doses. Letter SLNRC 86-1 (Reference 14) dated January 8, 1986, provided the required SGTR analysis. On November 12, 1986, the NRC issued a Request for Additional Information (RAI) (Reference 15) that raised concerns with the margin of 271 ft^3 (~5%) to steam generator overfill, which was originally calculated in the required SGTR analysis submitted by letter SLNRC 86-1. The RAI question stated, in part:

> The staff believes that, based on this response, there is sufficient uncertainty in the break flow calculations, as well as in operator action times (see Enclosure 2) and interval between safety injection (SI) termination and break flow termination (see question 2 below,) to warrant assuming that safety valve (SV) liquid relief can occur. …. Therefore please perform an analysis which assumes a design basis SGTR with loss of offsite power (LOOP), closure of the ruptured SG main steam isolation valve (MSIV), and SG overfill resulting in liquid relief through one SV.

In response to the RAI, WCNOC submitted letter WM 87-0145 (Reference 16) dated May 15, 1987, providing the results of a plant specific analysis that made changes to the modeling and assumptions for the SGTR analysis that would result in steam generator overfill and subsequent liquid relief through one safety valve. Subsequently, the NRC issued a safety evaluation (Reference 17) on May 7, 1991 that concluded that steam generator overfill does occur during a design basis SGTR accident and found the WCNOC SGTR accident analysis to be acceptable.

As part of the transition to the Westinghouse core design and safety analysis methodologies, the SGTR event was analyzed to demonstrate margin to overfill consistent with the Westinghouse standard analysis methodology. This analysis is presented in Section 2.7.2 of Enclosure I of this amendment request. The margin to overfill analysis is consistent with the methodology developed and approved in WCAP-10698-P-A (Reference 18) by Westinghouse. During the NRC review of the WCAP-10698-P-A methodology, the NRC indicated that an evaluation of the consequences of steam generator overfill would be desirable to demonstrate that the consequences of overfill would be acceptable. In response to this request, WCAP-11002 was submitted to the NRC (Reference 19). The NRC approval of WCAP-10698-P-A included an evaluation of WCAP-11002. The NRC approved methodology in WCAP-10698-P-A does not define a minimum margin to overfill requirement and does not require consideration of a forced overfill scenario.

ARCB-RAI-36

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

Response:

WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992, was transmitted to the NRC by Westinghouse Owners Group letter OG-92-25 on March 31, 1992, with the accompanying affidavit CAW-92-287. WCAP-13247, OG-92-25, CAW-92-287, and the NRC response to the Westinghouse Owners Group dated March 10, 1993 (which was an attachment to the Westinghouse Owners Group letter WOG-93-066) are included in Enclosure VIII of this LAR.

ARCB-RAI-37:

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

Response:

The RCS specific activity concentrations for each isotope is based on the assumed fuel defect level (1%), which is multiplied by the ratio of its dose conversion factor to the I-131 dose conversion factor for iodines and Xe-133 dose conversion factor for noble gases to obtain its I-131 or Xe-133 equivalence. The dose conversion factors used are the thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11 for iodines and the effective dose conversion factors from Table III.1 of EPA Federal Guidance Report No. 12, as identified in the revised Definitions for DOSE EQUIVALENT I-131 and revised DOSE EQUIVALENT XE-133 that are included in the Technical Specification markups contained in Enclosure IV, Section 9. The total of the isotopic dose equivalencies is the total I-131 or Xe-133 equivalence corresponding to the design fuel defect level of 1%. Each of the isotopic concentration values at the design defect level is then divided by the total I-131 or Xe-133 dose equivalence to obtain the isotopic concentrations at the applicable Technical Specification 3.4.16, "RCS Specific Activity," limits of $\leq$1.0 μCi/gm for DE I-131 and $\leq$500 μCi/gm for DE Xe-133.

Note that I-130, Kr-85, and Xe-131m are not listed in the Technical Specification Definitions for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. These nuclides do not contribute to the total of the isotopic dose equivalencies; however, their fuel defect level activities are divided by the total I-131 of Xe-133 dose equivalencies and scaled to the Technical Specification limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133.

ARCB-RAI-38

SRP 16.0 states, in part, that

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

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

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

Response:

As noted in the above question, SRP Section 16.0, "Technical Specifications," Rev. 3, Subsection II., "Acceptance Criteria," states, in part:

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

This subsection further states:

> Acceptable justifications for deviation would include retention of existing TS requirements, non-adoption of Standard Technical Specifications (STS) requirements not represented in existing TS (e.g., an LCO in STS but not in existing TS), editorial preference, facility design, and a technically justified alternative presentation equivalent to the STS intent. In some cases, comparison to the previous STS may help evaluate the proposed changes by clarifying the TS intent. The **previous** [emphasis added] STS NUREGs are as follows:
>
> - NUREG-0103, STS, Babcock and Wilcox Plants
> - NUREG-0452, STS, Westinghouse Plants
> - NUREG-0212, STS, Combustion Engineering Plants
> - NUREG-0123, STS, General Electric Plants

From the above, the quoted subsection specifies the TS change requests based on previous STS (i.e., NUREG-0452, not NUREG-1431) should comply with comparable provisions in these STS NUREGs (i.e., NUREG-1431) to the extent possible or justify deviations from the STS. The current WCGS TSs are based on NUREG-1431, Revision 1. WCNOC submitted a license amendment request on May 15, 1997 (Reference 6) that provided a conversion of the WCGS TSs that were based on NUREG-0452 to TSs that were based on NUREG-1431, Revision 1. The improved TSs were issued with the issuance of Amendment No. 123 on March 31, 1999 (Reference 7). The process for converting to the improved TSs requires justifications for deviations from the STS.

Subsection III of SRP 16.0, states, in part:

> The TS bases are not parts of the Operating License (OL) or Current Operating License (COL); however, the reference TS bases have a wealth of information on safety limit or LCO purposes and action and surveillance requirements. Particularly important is the bases description of how the system or parameter addressed by the LCO satisfies 10 CFR 50.36(c)(2)(ii). Regardless of any technical differences with the reference TS bases, the TS review should verify whether the plant–specific TS bases are consistent with the FSAR plant-specific accident analysis and system description. In addition, the TS bases should describe the basis for each TS requirement accurately.

Nuclear Reactor Regulation (NRR) Office Instruction No. LIC-100, Revision 1, "Control of Licensing Bases for Operating Reactors," Section 3.2, "Technical Specification Bases Section," states (page 3.10), in part:

> If the licensee and staff decide to process TS Bases pages, **even though NRC review and approval is not required** [emphasis added], the staff may "issue" the revised pages via a letter to the licensee (or as part of an amendment).

Section 2 of Enclosure VI of Reference 1 provided a description of the changes to the TSs and an associated justification for the change. The changes were further supported by the technical analysis in Section 4 of Enclosure VI of Reference 1.

Regarding the specific changes to TS Bases page B 3.6.3-2, these changes were made for consistency with the AST analyses and to correct a legacy issue associated with the current analyses of record for WCGS. The LOCA and rod ejection dose consequence analyses, which are the only two analyses that model isolation of containment, assumed the containment was isolated at the beginning of event, with the exception of the mini-purge releases for the first 10 seconds of the LOCA event. This is consistent with the current analyses of record for WCGS. 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.

Reference 4 RAIs

SRXB-RAI-1:

In the Summary of Analysis Codes Utilized in Postulated Accident Analyses, there are two Non-Loss-of-Coolant Accident (LOCA) Safety Analysis Codes listed for the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power, RETRAN, and LOFTRAN. Please provide the information that is obtained from each code and used as the basis for the updated analysis.

Response:

The departure from nucleate boiling (DNB) analysis of the rod withdrawal at power (RWAP) event is performed using the RETRAN code, consistent with the approved methodology in WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses." RETRAN is also the non-LOCA code used to perform DNB analyses for other Updated Safety Analysis Report (USAR) Chapter 15 events. The RWAP DNB cases performed consider 10%, 60%, and 100% power cases, at various reactivity insertion rates for minimum and maximum reactivity feedback conditions. Since the Revised Thermal Design Procedure (RTDP) DNB methods were used, initial condition uncertainties are accounted for in the DNB ratio (DNBR) safety analysis limit. As such, the initial conditions are set to their nominal values. A relatively small number of cases are performed in order to determine the most limiting set of conditions (e.g., a sensitivity study to determine conservative treatment of initial condition uncertainties is not required).

For RWAP cases analyzed to address RCS overpressurization concerns, the limiting set of conditions is established through varying a number of parameters (power, conservative treatment of initial condition uncertainties, reactivity insertion rates, etc.) resulting in a large number of cases to be considered. As discussed in WCAP-14882-P-A, the RETRAN and LOFTRAN codes are essentially equivalent. The LOFTRAN code is more suited for running a large number of cases than RETRAN; as such, it is the preferred transient analysis code for addressing this RCS pressure acceptance criterion. The LOFTRAN code (WCAP-7907-P-A, "LOFTRAN Code Description") has historically been the approved transient analysis code for non-LOCA analyses performed by Westinghouse and has previously been used to perform RWAP RCS overpressurization analyses for plants that otherwise utilize the RETRAN code for the DNB analyses. As a recent precedent, this approach was utilized for the Turkey Point Extended Power Uprate (Licensing Report Table 2.8.5.0-7: ADAMS Ascension Number ML103560177, and Safety Evaluation Report: ADAMS Ascension Number ML11293A365).

Enclosure I, Section 2.5.2.1.3 of this LAR has been updated to include the above discussion.

References:

1. WCNOC Letter ET 13-0023, “License Amendment Request for the Transition to Westinghouse Core Design and Safety Analysis,” August 13, 2013. ADAMS Accession No. ML13247A075.

2. Letter from C. F. Lyon, USNRC, to A. C. Heflin, WCNOC, “Wolf Creek Generating Station – Request for Additional Information Re: Transition to Westinghouse Core Design and Safety Analysis (TAC NO. MF2574),” April 3, 2014. ADAMS Accession No. ML14083A400.

3. WCNOC Letter ET 14-0017, “Withdrawal of License Amendment Request for the Transition to Westinghouse Core Design and Safety Analysis,” June 18, 2014. ADAMS Accession No. ML14175A119.

4. Letter from C. F. Lyon, USNRC, to A. C. Heflin, WCNOC, “Wolf Creek Generating Station – Request for Additional Information RE: Transition to Westinghouse Core Design and Safety Analysis (TAC NO. MF2574),” April 30, 2014. ADAMS Accession No. ML14111A100.

5. Letter from J. N. Donohew, USNRC, to O. L. Maynard, WCNOC, “Wolf Creek Generating Station – Issuance of Amendment Re: Elimination of Requirements for Post Accident Sampling Systems (TAC NO. MB0678),” March 2, 2001. ADAMS Accession No. ML010670233.

6. WCNOC letter ET 97-0050, “Technical Specification Conversion Application,” May 15, 1997. ADAMS Accession No. 9705210318 (Public Legacy Library).

7. Letter from J. N. Donohew, USNRC, to O. L. Maynard, WCNOC, “Conversion to Improved Technical Specification for Wolf Creek Generating Station – Amendment No. 123 to Facility Operating License No. NPF-42 (TAC NO. M98738),” March 31, 1999. ADAMS Accession No. ML022050061.

8. Letter from A. J. Mendiola, USNRC, to Technical Specifications Task Force, “Potential Issues with Plant-Specific Adoption of Travelers TSTF-51, Revision 2, “Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations,” TSTF-286, Revision 2, “Operations Involving Positive Reactivity Additions,” and TSTF-471, Revision 1, “Eliminate Use of Term Core Alterations in Actions and Notes”,” November 7, 2013. ADAMS Accession No. ML13246A358.

9. Not used.

10. Letter from J. N. Donohew, USNRC, to O. L. Maynard, WCNOC, “Wolf Creek Generating Station – Issuance of Amendment Re: Equipment Hatch Open During Refueling (TAC NO. MB2599),” July 30, 2002. ADAMS Accession No. ML022120428.

11. Letter from J. C. Stone, USNRC, to N. S. Carns, WCNOC, "Wolf Creek Generating Station – Amendment No. 95 to Facility Operating License No. NPF-42 (TAC NO. M94113)," February 28, 1996. ADAMS Accession No. ML022040718.

12. Letter from J. N. Donohew, USNRC, to O. L. Maynard, WCNOC, "Wolf Creek Generating Station – Issuance of Amendment Re: Use of Administrative Controls for Open Containment Penetrations During Refueling (TAC NO. MA9293)," September 12, 2000. ADAMS Accession No. ML003750021.

13. NUREG-0881, Supplement 5, "Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station, Unit No. 1," March 1985.

14. SNUPPS letter SLNRC 86-1, "Steam Generator Tube Rupture Analysis – SNUPPS," January 8, 1986.

15. Letter from P. W. O'Connor, USNRC, to G. L. Koester, Kansas Gas & Electric Company, "Request for Additional Information Related to the SNUPPS Steam Generator Tube Rupture Analysis," November 12, 1986.

16. WCNOC letter WM 87-0145, "Response to RAI Regarding SGTR Analysis," May 15, 1987.

17. Letter from D. V. Pickett, USNRC, to B. D. Withers, WCNOC, "Safety Evaluation Report for the Wolf Creek Generating Station Steam Generator Tube Rupture Analysis (TAC NO. 57363)," May 7, 1991.

18. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," August 1987.

19. WCAP-11002 (Proprietary) and WCAP-11003 (Non-Proprietary), "Evaluation of Steam Generator Overfill Due to a SGTR Accident," February 1986.

20. WCNOC letter ET 14-0003, "Response to Request for Additional Information Regarding License Amendment Request for the Transition to Westinghouse Core Design and Safety Analysis," January 28, 2014. ADAMS Accession No. ML14035A224.

21. WCNOC letter WO 14-0031, "Response to Request for Additional Information Regarding License Amendment Request for the Transition to Westinghouse Core Design and Safety Analysis," March 20, 2014. ADAMS Accession No. ML14091A245.

22. WCNOC letter WO 14-0032, "Supplemental Information for Response to Request for Additional Information Regarding License Amendment Request for the Transition to Westinghouse Core Design and Safety Analysis," March 26, 2014. ADAMS Accession No. ML14091A261.

23. Letter from R. F. Kuntz, USNRC, to C. M. Crane, Exelon Generation Company, "Byron Station, Unit Nos. 1 and 2, and Braidwood Station, Unit Nos. 1 and 2 – Issuance of Amendment Re: Alternative Source Term (TAC NOS. MC6221, MC6222, MC6223, and MC6224)," September 8, 2006. ADAMS Accession No. ML062340420.

24. Letter from C. F. Lyon, USNRC, to M. W. Sunseri, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Re: Transition to Westinghouse Core Design and Safety Analysis (TAC NO. MF2574)," December 13, 2013. ADAMS Accession No. ML13345B335.

25. Letter from C. F. Lyon, USNRC, to A. C. Heflin, WCNOC, "Wolf Creek Generating Station – Request for Additional Information Re: Transition to Westinghouse Core Design and Safety Analysis (TAC NO. MF2574)," March 5, 2014. ADAMS Accession No. ML14058A088.

26. Email from Fred Lyon, USNRC, to Steve G Wideman, WCNOC, "FW: Wolf Creek Generating Station - Request for Additional Information Re: Transition to Westinghouse Core Design and Safety Analysis (TAC No. MF2574)," December 23, 2013. ADAMS Accession No. ML13357A250.

27. Letter from C. F. Lyon, USNRC, to M. W. Sunseri, WCNOC, "WOLF CREEK GENERATING STATION- REQUEST FOR ADDITIONAL INFORMATION RE: TRANSITION TO WESTINGHOUSE CORE DESIGN AND SAFETY ANALYSIS (TAC NO. MF2574)," January 28, 2014. ADAMS Accession No. ML14027162.