Attachment 1 to SAP-18-34, "Supplemental 30 Day Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies"
Non-Proprietary

Supplemental 30 Day Responses to Nuclear Regulatory Commission Request for Additional Information Regarding Wolf Creek Generating Station Transition to Westinghouse Safety Analysis and Alternate Source Term Methodologies [Non-Proprietary]

(15 pages including cover page)

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RAI ARCB1-FHA-2 - FHA

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

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

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

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

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

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

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

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

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

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

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

Response

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

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

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

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

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

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

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

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

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

Where

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

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

$DF_{elemental}$ is the elemental iodine DF

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

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

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

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

Supplemental Response

As a result of the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office, additional insights are provided to support the assertion that the overall iodine DF of 200 remains conservative at fuel pin pressures of up to 1500 psig.

The formulation used within the paper by G. Burley, "Evaluation of Fission Product Release and Transport for a Fuel Handling Accident," 1971, ADAMS Accession No. ML16357A003 (henceforth referred to as the "Burley paper") can be used to provide additional support that the overall iodine DF of 200 remains conservative at fuel pin pressures of up to 1500 psig.

The Burley paper formula for the inorganic (i.e. elemental) iodine DF is:

$$\text{DF inorg} = e^{\left(6/d_b\right)\left(k_{eff}\right)\left(H/v_b\right)}$$

Where

db = bubble diameter (cm)

vb = bubble rise velocity (cm/sec)

H = bubble rise height (cm)

keff is calculated:

$$k_{eff} = \frac{1}{\left(\frac{1}{k_o + k_E}\right) + \left(\frac{1}{k_L P}\right)}$$

Where

ko = 1.646*DG/db

DG = Diffusivity of Iodine in He, 0.278 cm2/sec

kE = 3.75E-3 vb (turbulent flow)

kL = 1.13(DLvb/db)0.5

DL = Diffusivity of Iodine in water, 1.27E-5 cm2/sec

P = Iodine partition factor, 10 (selected based on Section VIII of the Burley paper).

The bubble rise height, H, is 23 feet, or 701.04 cm. The bubble rise time used in the Burley paper is 4.7 seconds; this rise time is based on data taken from WCAP-7518-L, Table 3-5 and corresponds to a release pressure of 1200 psig. With a the rise height of 701.04 cm and the rise time of 4.7 seconds, a bubble rise velocity, vb, of 149.16 cm/sec is calculated. Based on the Burley paper formula, increases in either bubble diameter or bubble rise velocity would result in lower calculated DFs.

RG 1.183, Appendix B, Position 2, states, in part, "...the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200..." Using the calculated bubble rise velocity of 149.16 cm/sec and a rise height of 701.04 cm, an elemental iodine DF of approximately 500 results from a bubble diameter of approximately 1.319 cm. This bubble diameter is used in further calculations described below.

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These insights provide additional assurances that the overall DF of 200 remains conservative at fuel pin pressures of up to 1500 psig.

RAI ARCB1-SGTR-2 - SGTR

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

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

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

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

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

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

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

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

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

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

Response

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

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

(ADAMS Accession No. ML012830516, Section 3.1.2), and Diablo Canyon (ADAMS Accession No. ML17012A246, Section 3.3.7).

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

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

Supplemental Response

As a result of the audit conducted on March 19 and 20, 2018 at the Westinghouse Rockville, MD office, additional discussion is provided for the SGTR scenario when considering a loss of offsite power at the start of the event; the additional discussion is related to the modeling of the control room, and includes a summary of the control room modeling assumptions. For completeness, the other events are also included in the summary of control room modeling assumptions.

The control room isolation is modeled in the dose analyses in two parts: actuation of the emergency mode filtration (in both the control building and the control room) and closure of the normal HVAC intake damper. The actuation of the emergency mode filtration occurs following receipt of an isolation signal (e.g. high radiation or safety injection). Closure of the normal HVAC intake damper occurs on a safety injection signal or manual action. Prior to closure of the normal HVAC intake damper, the unfiltered inleakage is modeled with a X/Q consistent with the normal mode HVAC intake; after this occurrence, the unfiltered inleakage is modeled with a X/Q consistent with the emergency mode HVAC intake. In the analyses, a failure of one of the filtration fans is assumed at the start of emergency mode resulting in a larger unfiltered inflow to the control room (since only half of the makeup flow to the control room passes through a filter). After a defined time of 90 minutes from the start of the event, operator action isolates the failed train and terminates the unfiltered inflow to the control room, and consequently lowers the filtered inflow to the control building.

Following the SGTR, the releases of noble gases results in an immediate high radiation signal; noble gases have a condenser DF of 1 and so are not affected by loss of offsite power timing. Emergency mode filtration would be actuated following a 60-second delay to account for instrumentation delays and damper movement. The SGTR dose analysis arbitrarily and conservatively models emergency mode filtration as being actuated at 120 seconds.

The safety injection signal is generated at 325 sec after event initiation and normal HVAC intake damper closure would occur at 385 sec after event initiation (including a 60 second delay for instrumentation delays and damper movement). The SGTR analysis conservatively models normal HVAC intake damper closure at 10 minutes (600 seconds) after event initiation.

With respect to the SGTR, a loss of offsite power at the start of the event would not significantly affect the plant response, only the timing. The major dose contributors, such as post-trip flashed break flow, total break flow and post-trip ruptured SG releases are driven by decay heat removal and so would be unaffected. Essentially, the sequence of events with respect to the releases would shift by approximately

52 seconds (0.014 hours); these 52 seconds are also within the conservatism added to the start of emergency mode filtration and normal HVAC intake damper closure. Removing the first 52 seconds of releases from the model and removing added conservatism to the control room emergency mode actuation and normal HVAC intake damper closure would result in a slight decrease in calculated dose.

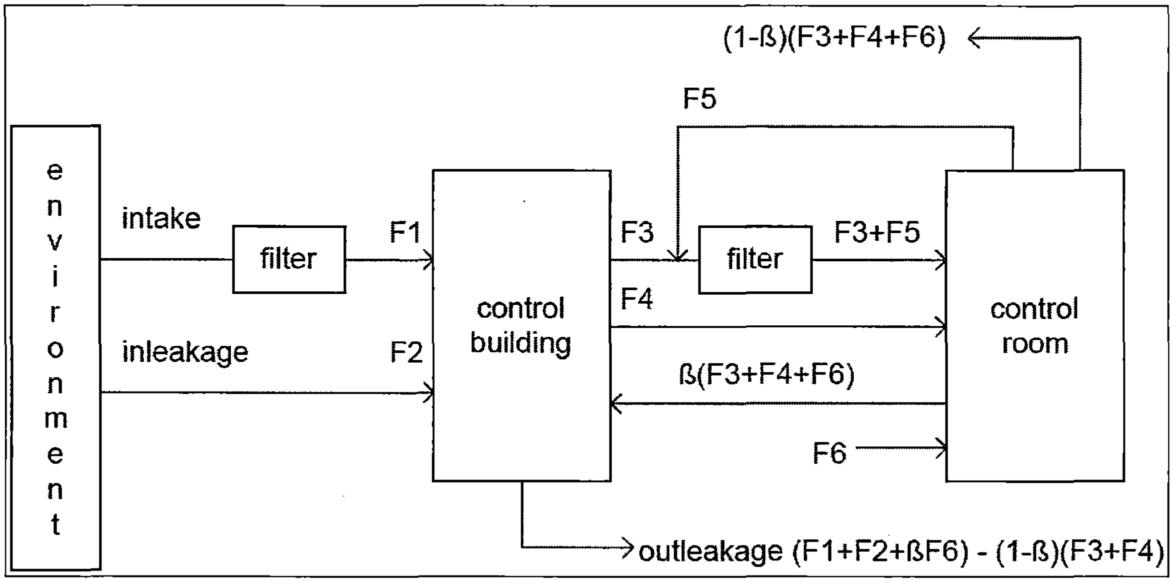
A summary of control room modeling assumptions for all events is summarized below:

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

*See response to ARCB1-MSLB-2

The HVAC flows modeled are as follows:

Control Room and Control Building Ventilation Flows



Control Room and Control Building Ventilation Flows

Flow Path	Normal Mode Flow (cfm)	Emergency Mode Flow Prior to Operator Action (cfm)	Emergency Mode Flow After Operator Action (cfm)
F1	0	1350	675
F2	13050	400	400
F3	0	550	550
F4	0	550	0
F5	0	1250	1250
F6	2000*	50	50
β	0	0	0

* 1950 cfm makeup + 50 cfm inleakage

RAI ARCB1-MSLB-2 - MSLB

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

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

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

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

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

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

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

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

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

1. Please state the assumed time for the SI setpoint to be reached and state the reference analysis used to determine this value. Justify how assuming this time results in the worst case radiological

consequences and why the SI signal is credited when the USAR says there are conditions when it would not be available.

Response

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

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

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

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

Section 4.3.3.2.3 would be revised to read as follows:

4.3.3.2.3 Control Room

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

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

4.3.3.4 Results and Conclusions

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

For the pre-accident iodine spike case:

- Control room 4.5 rem TEDE

For the accident-initiated iodine spike case:

- Control room 4.8 rem TEDE

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

Table 4.3-6 Assumptions Used for Main Steamline Break Analysis

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

Supplemental Response

The third sentence of the first paragraph is revised (revisions in italics) to "Relative to the completion of the faulted SG blowdown, earlier control room isolation reduces the activity ingress due to filtration and later control room isolation reduces the activity *in the control room by exhausting it from* the control room."

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

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

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

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

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

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

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

Response:

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

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

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

Supplemental Response

The original response to this question incorrectly referred to RAI ARCB1-WT-4 when discussing the partitioning of the iodine in the VCT. The correct reference should have been to RAI ARCB1-WT-5. So the correct wording will be:

"…Partitioning of the iodine in the VCT is conservatively addressed as explained in the response to ARCB1-WT-5 by using a total partition coefficient of 100 for the RCS chemical species split of 95% particulate, 4.85% elemental, and 0.15% organic…"

Enclosure II to ET 18-0012

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