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W3F1-2004-0095

October 13, 2004

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

SUBJECT: Supplement 3 to Amendment Request NPF-38-256,
Alternate Source Term
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. Entergy Letter dated July 15, 2004, "License Amendment Request NPF-38-256, Alternate Source Term"
2. Entergy Letter dated August 19, 2004, "License Amendment Request NPF-38-256, Supplement to Alternate Source Term Submittal"
3. Entergy Letter dated September 1, 2004, "Supplement 2 to Amendment Request NPF-38-256, Alternate Source Term"
4. NRC Letter dated September 29, 2004, "Waterford Steam Electric Station, Unit 3 (Waterford 3) - Request for Additional Information Related to Revision to Facility Operating License and Technical Specifications - Extended Power Uprate Request (TAC No. MC1355) and Alternate Source Term Request (MC3789)"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) licensing basis to implement an Alternate Source Term (AST) for calculating accident offsite doses and doses to control room personnel as permitted by 10 CFR 50.67. Entergy supplemented this request via References 2 and 3.

By letter (Reference 4), the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI) regarding the AST submittal. Responses to the RAI are provided in Attachment 1.

The original no significant hazards consideration included in Reference 1 is not affected by any information contained in the supplemental letter. This submittal includes new commitments as summarized in Attachment 2.

If you have any questions or require additional information, please contact Jerry Burford at 601-368-5755.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 13, 2004.

Sincerely,

KJP/DBM/cbh

Attachments:
1. Response to Request for Additional Information
2. List of Regulatory Commitments

cc: Dr. Bruce S. Mallett
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Attachment 1

W3F1-2004-0095

Response to Request for Additional Information

Response to Request for Additional Information

Question 1:

How were the values in Table 1-1.A, "Core Inventory for Steaming Events," determined?

Response 1:

The core inventory tables used in the AST analyses, Table 1-1 and 1-1.A, are discussed in Section 1.2 of the W3F1-2004-0053 Licensing Report (pp. 4-5). The Table 1-1.A inventory is the one described as being generated using the ORIGEN2 code to determine the gap fission product activities in peak power rods. This inventory was used in the dose calculations for events with releases via Atmospheric Dump Valves (ADVs) and/or Main Steam Safety Valves (MSSVs).

The Table 1-1.A isotope inventories were generated based on depletion of fresh fuel rods, capturing the worst case fission product activities associated with burnups through 40,000 MWD/T. Fuel rods beyond this burnup are not capable of full power operation and thus have non-limiting inventories. Table 1-1.A focused on gas gap activities, thus provided inventories for Iodines and Noble Gas only. In contrast, the Table 1-1 inventories were generated based on an equilibrium power uprate core design, and thus accounted for different batch burnups at assumed Beginning of Cycle (BOC). Worst case time in cycle inventories for each isotope were determined and used as the basis for this core inventory. This resulted in higher inventories for several Krypton isotopes and Xe-135 in Table 1-1.A since those isotopes have higher inventories at BOC compared to Middle of Cycle (MOC).

Question 2:

Table 1-2 lists the secondary coolant mass for two conditions.

A. Are these mass values per steam generator (SG) or the total mass for both SGs?
B. Is this the liquid mass only, or does it include the mass of the steam in the secondary system?
C. What values are used for secondary coolant mass in each SG for each of the design basis accident (DBA) dose analyses that assume a steaming release from the secondary coolant system?

Response 2:

The assumed secondary inventory values of 153,700 lbm at Hot Full Power (HFP) and 241,450 lbm at Hot Zero Power (HZP) are values of the liquid mass for a single SG. The larger HZP value was considered more representative of the SG mass present in an intact SG that is being cooled to Shutdown Cooling (SDC) entry conditions, and maximizes the initial activity present for the case of releases from a faulted SG. Therefore, the HZP values were used as the basis for all of the release calculations except for Steam Generator Tube Rupture (SGTR). For the SGTR event, a low initial SG level (corresponding to 106,300 lbm per SG) was conservatively assumed since this results in earlier uncovery of the top of the SG tubes, which results in increased flashing and a lower decontamination factor. This is a very conservative value that corresponds approximately to the reactor trip setpoint on Steam Generator Level Low.

Question 3:

What is the assumed value of the reactor coolant system mass for each of the DBA dose analyses that assume primary to secondary leakage and a steaming release from the secondary coolant system?

Response 3:

Except for SGTR, the dose analyses assuming primary-to-secondary leakage assume the following Reactor Coolant System (RCS) masses:

34,260	Pressurizer Liquid Mass (lbm)
4,016	Pressurizer Steam (lbm)
395,502	Non-Pressurizer Liquid Mass (lbm)

For purposes of determining activity concentration, only the non-pressurizer liquid mass is considered. For purposes of determining the steaming release due to cooldown, the liquid and steam masses of the pressurizer are also considered.

For SGTR, the dose analyses are based on the detailed CENTS analysis performed in support of the information presented in Section 2.13.6.3.2 of the Extended Power Uprate (EPU) report submitted via W3F1-2003-0074 dated November 13, 2003. The total initial RCS mass of 467,000 lb is assumed for that analysis.

Question 4:

The DBA control room habitability analyses take credit for the operators manually selecting, at 2 hours, the intake with the lesser amount of radioactivity entering, as per NUREG-0800, "Standard Review Plan," (SRP) Section 6.4. This credit uses the atmospheric dispersion factor (X/Q) for the more favorable intake reduced by a factor of 4, and is subject to some limitations as discussed in SRP 6.4 and Regulatory Guide (RG) 1.194.

A. Are the two control room intakes in different wind direction windows?
B. Are there redundant, engineered safety feature-grade radiation monitors within each intake with control room indication and alarm?
C. Are there procedures and training to direct the control room operators to select the least contaminated outside air intake and to take steps to monitor to ensure the least contaminated intake is in use throughout the event?

Response 4:

The two control room intakes meet the criteria for being in different wind direction windows, and are equipped with redundant, safety grade radiation monitors within each intake that provide alarms and indications to the Main Control Room (MCR). The existing off-normal operating procedure for High Airborne Activity in the Control Room provides directions for establishing pressurization flow, using outside air from the intake with the lowest activity readings. Procedures and training related to these actions are being reviewed for revisions and enhancements to be implemented prior to implementing the 3716 MWt EPU.

Question 5:

What is the basis for the assumed reduction in the sprayed fraction of containment from 0.85 to 0.80? Is this change based on a revised analysis of the sprayed volume of containment?

Response 5:

Waterford 3 current licensing basis analyses are based upon a 0.85 sprayed volume fraction. This value had been reviewed several years ago and it was determined that the sprayed volume fraction was actually slightly less than 0.85, although it was judged that there were sufficient conservatisms present in the Loss of Coolant Accident (LOCA) analysis that it was not necessary to revise the analyses to account for this. It was decided, as an enhancement, that a more conservative value of 0.80 would be specified for the AST Large Break (LBLOCA) analyses.

Question 6:

What is the basis for the assumed removal coefficient for natural deposition of elemental iodine of 0.4 per hour?

Response 6:

Waterford 3 models natural deposition of elemental iodine per SRP, Section 6.5.2, where the coefficient is given by:

$$\lambda_w = K_w A / V$$

A containment volume, V, of 2.677E+06 ft^3 is assumed. The surface area available, A, is conservatively estimated as that corresponding to the cylindrical structure of the containment, with a 70 foot radius and a 150 foot height, for an area of about 66,000 ft^2. A K_w value of 4.9 meters/hour is assumed. This results in a λ_w for elemental iodine of 0.40/hour.

Question 7:

For the large break LOCA (LBLOCA) emergency core cooling system (ECCS) leakage release pathway analysis, what input value is assumed for the sump volume or mass? What is the basis for this value?

Response 7:

A minimum sump liquid volume of 57,900 ft^3 is assumed to maximize the activity concentration in evaluating the impact of the Engineered Safety Feature (ESF) liquid leakage contribution in the LBLOCA analysis. It corresponds to assuming minimum available initial volumes for the Refueling Water Storage Pool, Safety Injection Tanks, etc., to maximize radioiodine concentrations in sump water that leaks into the Reactor Auxiliary Building. This value contrasts to a maximum sump liquid volume of 108,771 ft^3 which is assumed for the Reactor Building air leakage pathway contribution to the LBLOCA analysis.

Question 8:

The small break LOCA (SBLOCA) secondary containment steaming pathway analysis assumes a reduced primary-to-secondary leakage at the Technical Specification (TS) limit of 75 gallons per day (gpd), as proposed for the extended power uprate (EPU) amendment request (TAC MC1355). The current TS limits are 1 gallon per minute (gpm) total primary to secondary and 720 gpd through any one SG. This amounts to almost a factor of 10 reduction in the allowed leakage. Does the 75 gpd primary-to-secondary leakage assumed in the dose analysis bound the expected leakage due to the SBLOCA?

Response 8:

Primary-to-secondary leakage assumptions for the AST dose analyses are discussed in Section 1.2 of the W3F1-2004-0053 Licensing Report. Specifics of this assumption with respect to SBLOCA are also discussed in Section 6.1.3 of the report. The SBLOCA event is considered an event, similar to Control Element Assembly (CEA) Ejection, for which SGs are in a non-faulted condition. Because there will be a smaller differential pressure across the SG tubes under SBLOCA conditions than under normal operating conditions, it is reasonable to assume the 75 gpd limit that will be applied to normal operating conditions is bounding for SBLOCA conditions.

Question 9:

The inside containment MSLB analysis assumes primary-to-secondary leakage of 540 gpd through the faulted SG and 150 gpd for the unaffected SG for this accident, whereas the newest proposed TS limit for the EPU submittal is 75 gpd primary-to-secondary leakage through any steam generator.

A. What is the basis for the faulted SG leakage value of 540 gpd?
B. What amount of leakage could be expected through the SG tubes on the affected SG for a postulated MSLB inside containment? Does the 540 gpd primary-to-secondary leakage assumed in the dose analysis bound the expected leakage?
C. Considering that the calculated control room dose is fairly close to the limit, how much primary-to-secondary leakage can be tolerated for this accident without going over the 10 CFR 50.67 and General Design Criterion (GDC)-19 total effective dose equivalent (TEDE) limit of 5 rem in the control room?

Response 9:

Primary-to-secondary leakage assumptions for the AST dose analyses are discussed in Section 1.2 of the W3F1-2004-0053 Licensing Report. As stated in Section 7.1.3 of that report, a value of 540 gpd is assumed for the affected SG and a value of 150 gpd is assumed for the intact SG. Because the affected SG blows down to containment, the results of the Inside Containment MSLB analysis are relatively insensitive to its assumed 540 gpd primary-to-secondary leak rate.

The amount of leakage that could be expected through the affected SG during a MSLB will be less than the Accident Leakage amount. Therefore, the 540 gpd assumed primary-to-secondary leakage does bound the "expected leakage". The 540 gpd accident induced primary-to-secondary leak rate assumed under accident conditions for faulted SG conditions will be protected by the Steam Generator Operational Assessment performed per NEI 97-06 for each operating cycle.

<u>Background</u>
All PWR Utilities are committed to the NEI 97-06 Performance Criteria. The three criteria are:

1. Structural Integrity
2. Accident Induced Leakage Criteria
3. Operational Leakage Criteria

Compliance with the performance criteria is accomplished through a combination of SG examination (ISI), analysis (SG Integrity Assessments) and on-line leakrate monitoring.

Structural Integrity and Accident Leakage

Waterford 3 performs SG testing in accordance with NEI 97-06, "Steam Generator Program Guidelines," Revision 1. Section 3.1.3 of that document requires that a Tube Integrity Assessment be performed after each SG inspection (which occurs each refueling outage for Waterford 3). The purpose of the integrity assessment is to ensure that performance monitoring criteria have been met for the previous operating period (i.e., condition monitoring), and will continue to be met for the next period (i.e., operational assessment). As part of the operational assessment Waterford 3 evaluates the current states of each SG and performs conservative analyses to predict potential crack growth to demonstrate that the structural integrity performance criterion is not exceeded. Specifically, SG tubing must retain structural integrity over the full range of normal operations and design basis accidents. This includes a safety factor of 3.0 against burst under normal steady state full power operation, and a safety factor of 1.4 against burst under the limiting design basis accident (i.e., a MSLB).

The guidelines further require that the operational assessment be completed within 90 days after startup. Should this assessment conclude that the criteria set forth by the site safety analysis could potentially be exceeded, corrective actions would immediately be implemented to ensure that this could not occur. This could result in the requirement of additional inspections (i.e., a "mid-cycle" outage for SG tube inspections), or potential revisions to the site dose analyses. Thus, the Operational Assessment and the Waterford 3 commitment to NEI 97-06 ensures that the 540 gpd value will adequately characterize the SG primary-to-secondary leakage rate under MSLB accident conditions.

Waterford 3 reduced the assumed accident induced SG primary-to-secondary leakage through a failed SG from the current value of 720 gpd to 540 gpd for EPU and AST. The value was reduced to ensure that the dose criteria set forth by 10CFR100 and 10CFR50.67 are met. Both the impact to dose consequences and current plant conditions were considered in determining this value.

Operational Leakage Criteria

Operational Leakage Criteria is met through the on-line leakage monitoring program. Waterford 3 committed to adopting an operational leakage limit as the TS limit for SG primary-to-secondary leakage in letter W3F1-2004-0017, dated March 4, 2004. This was in response to discussions with the NRC related to NRC RAI Question 5 of NRC letter dated January 28, 2004. The 540 gpd accident induced primary-to-secondary leak rate assumed under accident conditions for faulted SG conditions will be protected by the Steam Generator Operational Assessment performed per NEI 97-06 for each operating cycle.

The AST Licensing Report documented a MCR dose of 4.888 Rem TEDE for the Inside Containment MSLB, based upon a 150 gpd primary-to-secondary leak rate for the intact SG and a constant 100 CFM unfiltered in-leakage through the event. This is less than the 5 Rem TEDE acceptance limit of 10CFR50.67. Note Waterford 3 has proposed to adopt a 75 gpd per SG TS limit on SG operational leakage, and that no credit is taken for reduced unfiltered in-leakage when the Control Room is assumed to be pressurized. Additionally, the analysis conservatively assumes that the Control Room is pressurized from the start of the event. Further, operator action to select the preferred (lower radiation level) control room air intake is only postulated at two hours into the event. Thus, while the documented results are close to the acceptance limits, the analysis has been conservatively constructed to allow operators flexibility in the timing of their actions. This results in inherent margins within the calculation.

A sensitivity case was performed in the Inside Containment MSLB calculation to determine the effects of primary-to-secondary leakage. With all other parameters unchanged, for a 75 gpd per SG leak rate for the intact SG the resultant dose was 2.766 Rem TEDE, or 56.6% of the 4.888 Rem TEDE value corresponding to a 150 gpd leak rate. Thus, the TEDE dose results are approximately linear with assumed primary-to-secondary leakage. By extrapolation, the 5 Rem TEDE limit would be reached for a primary-to-secondary leak rate of 153.9 gpd.

Question 10:

On page 44 of the submittal, the timeline for the SGTR accident indicates that the operator would open the atmospheric dump valve (ADV) on the affected SG as needed after 6.6 hours. This later steaming release is not accounted for in the dose analyses of the SGTR accident. Revise the analysis of the SGTR to include the release from the affected SG ADV after 6 hours.

Response 10:

Interpretation of RG 1.183 requirements on releases for SGTR was discussed at the August 12, 2004, meeting between Entergy and NRC. As a result of these discussions, Entergy agrees to revise the analysis as requested.

Question 11:

On pages 48 and 49 of the submittal, the calculation of scaled effective control room X/Qs is discussed. The table on page 49 includes the base control room X/Qs used in the calculation.

A. The 2-8 hr X/Q for "SG_1 to MCR" [main control room] is not the same value as appears in the table in the middle of page 48 for "East ADV to West MCR Air Intake," which the staff assumes is the same source-receptor pair. Should these be the same value?
B. The 2-8 hr X/Q for "SG_2 to MCR" is not the same value as appears in the table in the middle of page 48 for "West ADV to West MCR Air Intake," which the staff assumes is the same source-receptor pair. It instead appears to be unchanged from the 0-2 hr "SG_2 to MCR" X/Q value. Although it results in a conservative dose, why was this X/Q unchanged?

Response 11:

The 2-8 hour X/Q value for SG_2 to MCR should be the .00562 value per Table 1-3. This was an inadvertent minor conservative error which has been corrected in the most recent SGTR analysis. Similarly, the 2-8 hour value for SG_1 releases to the MCR in the table did not account for the factor of 4 reduction for selection of the preferred control room air intake for pressurization flow. The weighting values will change as a result of the ongoing reanalysis (see Response 10) to fully account for early releases from the affected SG using both ADVs for the early rapid cooldown prior to isolation. This issue has been entered into Entergy's 10 CFR 50 Appendix B corrective action process at Waterford 3.

Question 12:

Provide the calculated value for the control room dose due to the SGTR with accident-induced iodine spiking.

Response 12:

This information will be provided in conjunction with the information requested in Question 10.

Question 13:

The analysis of the MSLB outside containment assumes increased primary-to-secondary leakage of 540 gpd through the faulted SG and 150 gpd for the unaffected SG for this accident, whereas the newest proposed TS limit for the EPU submittal is 75 gpd primary-to-secondary leakage through any SG.

A. What is the basis for the faulted SG leakage value of 540 gpd?
B. What amount of leakage could be expected through the SG tubes on the affected SG for a postulated MSLB outside containment? Does the 540 gpd primary-to-secondary leakage assumed in the dose analysis bound the expected leakage?
C. Considering that the calculated control room dose is fairly close to the limit, how much primary-to-secondary leakage can be tolerated for this accident without going over the 10 CFR 50.67 and GDC-19 TEDE limit of 5 rem in the control room?

Response 13:

Primary-to-secondary leakage assumptions for the AST dose analyses are discussed in Section 1.2 of the W3F1-2004-0053 Licensing Report. As stated in Section 9.1.3 of that report, a value of 540 gpd is assumed for the affected SG and a value of 150 gpd is assumed for the intact SG for the Feedwater Line Break / Outside Containment MSLB event.

As discussed in the response to Question 9 above, the Waterford 3 Steam Generator Operational Assessment and the Waterford 3 commitment to NEI 97-06 ensures that the 540 gpd value will adequately characterize the SG primary-to-secondary leakage rate under MSLB accident conditions.

For this event, the AST Licensing Report documented an MCR TEDE dose of 3.6158 Rem TEDE for the Accident Generated Iodine Spike (GIS) case and of 2.2029 Rem TEDE for the Pre-existing Iodine Spike (PIS) case. The affected SG is the dominant contributor to the dose for this event. Although explicit parametric studies for the affect of primary-to-secondary leakage assumptions were not performed for this event, engineering judgment based on other events (CEA Ejection, Inside Containment MSLB) lead to the conclusion that the MCR TEDE dose contribution is slightly less than linear with increasing primary-to-secondary leakage. Thus, assuming all of the reported MCR dose is due to the affected SG, by linear extrapolation, a 5 Rem TEDE MCR dose would result for the GIS case based on a 847 gpd (0.588 gpm) leak rate for the affected SG; a 5 Rem TEDE MCR dose would result for the PIS case based on a 1,226 gpd (0.85 gpm) leak rate.

Note no credit is taken for reduced unfiltered in-leakage when the Control Room is assumed to be pressurized. Additionally, the analysis conservatively assumes that the Control Room is pressurized from the start of the event. Further, operator action to select the preferred (lower radiation level) control room air intake is only postulated at two hours into the event. Thus the analysis has been conservatively constructed to allow operators flexibility in the timing of their actions. This results in inherent margins within the calculation.

Question 14:

The LBLOCA and the LBLOCA shine analyses both have a reduced assumption for ECCS leakage of 0.5 gpm, which was previously 1 gpm. What is the basis for this change?

Response 14:

Waterford 3 elected to reduce the assumed value for ECCS leakage in the LBLOCA RADTRAD analysis for offsite dose and control room dose and in the LBLOCA shine to meet the GDC19 criterion of 5 Rem TEDE dose to occupants in the control room envelope. As stated in W3F1-2004-0076, dated Sept. 1, 2004, Waterford 3 has committed to revising its plant procedures to specify a maximum leakage of one half of the value specified in LOCA radiological analyses. These procedures currently specify a 1.0 gpm value; this will be reduced to a 0.25 gpm value, consistent with RG 1.183, to support the assumption of a 0.5 gpm value. Review of past surveillance data indicates margin exists to accommodate this change.

Question 15:

The questions above (5, 6, and 7) for the July 15, 2004 AST submittal LBLOCA analysis still apply to the revised LBLOCA analysis.

Response 15:

Questions 5-7 are addressed based on the revised LBLOCA analysis reported in W3F1-2004-0078. There has been no change in the modelling associated with those parameters.

Question 16:

With regard to the RADTRAD calculations performed to provide source term input for the LBLOCA shine calculations in MicroShield, besides the changes noted in Section 5 of the September 1, 2004, supplement, are there any other differences as compared to the DBA LBLOCA containment release pathway assumptions, as discussed in Section 4 of that supplement?

Response 16:

Yes. The release pathways with potential for fission product deposition onto filters presented for the DBA LBLOCA and the filter shine calculations considered fission product releases from containment via four mechanisms:

- Normal Containment Air Leakage directly to the environment,
- Normal Containment Air Leakage to the Shield Building Annulus,
- Normal Containment Air Leakage to the Controlled Ventilation Area System (CVAS), and
- ECCS fluid leakage.

The RADTRAD DBA LBLOCA model for normal containment air leakage considered a leak rate of 0.5% volume per day for the first 24 hours of the accident and then 0.25% volume per day for the duration of the accident. The normal containment air leakage is distributed assuming 40% is released to the Shield Building Annulus, 6% directly to the environment, and 54% directly to the CVAS. The RADTRAD DBA LBLOCA model also assumed a control room unfiltered in-leakage rate of 100 CFM and the filtered release paths assumed 99% filter efficiencies.

The LBLOCA RADTRAD model used to calculate the fission product loading on the Control Room Recirculation Filters assumed that the 6% direct containment leakage was deposited on the Control Room Recirculation Filters based on a conservative control room unfiltered in-leakage rate of 200 CFM and 100% filter efficiencies. The 40% leakage that was directed to the Shield Building Annulus was assumed to be deposited on the Shield Building Ventilation System (SBVS) filters. The LBLOCA RADTRAD model used to calculate the fission product loading on the CVAS filters conservatively assumed 60% of the normal containment air leakage (54% directed to the CVAS + the 6% Direct Bypass to the environment) was deposited on the CVAS filters.

The RADTRAD DBA LBLOCA ECCS fluid leakage pathway assumed a leak rate of 0.5 gpm (2 times the allowable leak rate per RG 1.183) with all the iodine assumed deposited in the sump water. This model also assumed a constant flashing fraction of 10% for the entire 30-day duration of the accident. The RADTRAD ECCS fluid leakage pathway model for the filter loadings for the shielding calculation was done using the same leak rate and initial inventory of iodine in the sump water, however a reduced flashing fraction after 24 hours was credited (see Response to Question #18 for justification).

Question 17:

What is the basis for assuming a reduced flashing fraction in the ECCS leakage pathway analysis for the shine dose source term calculations, as compared to the analysis performed to determine the inhalation and submersion dose for the LBLOCA?

Response 17:

The DBA LBLOCA analysis performed to calculate the inhalation and submersion main control room dose assumed a flashing fraction of 10% for the entire 30-day duration of the accident; however the RADTRAD calculations for the filter shine calculations assumed a reduced flashing fraction of 2% after 24 hours.

See the response to Question #18 for the justification for assuming a reduced flashing fraction after 24 hours for the LBLOCA RADTRAD results used to determine the filter loading on the CVAS filters for LBLOCA source term due to the ECCS leakage pathway.

Question 18:

Page 12 of the September 1, 2004, supplement to the AST amendment request provides a constant enthalpy calculation of the maximum flashing fraction, based on the maximum ECCS fluid temperature. The filter shine dose analyses assumed a flashing fraction value of 2% based on the result of this calculation multiplied by a factor of 10. Section 5.5 of Appendix A of RG 1.183 states that for leakage with temperatures less than 212 °F or for calculated flashing fractions less than 10%, the airborne iodine should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates. Provide the justification for the lower flashing fraction value based on the actual sump pH history and area ventilation rates. Consider also the projected pH of the ECCS leakage and area ventilation rates for the DBA LBLOCA.

Response 18:

The Waterford 3 AST assessment for radioactive filter shine assumes a flashing fraction of 10% for the first 24 hours of the accident and then reduces the flashing fraction to a value of 2% based on the result of the constant enthalpy calculation of the maximum flashing fraction based on the maximum ECCS fluid temperature multiplied by a factor of 10 for conservatism.

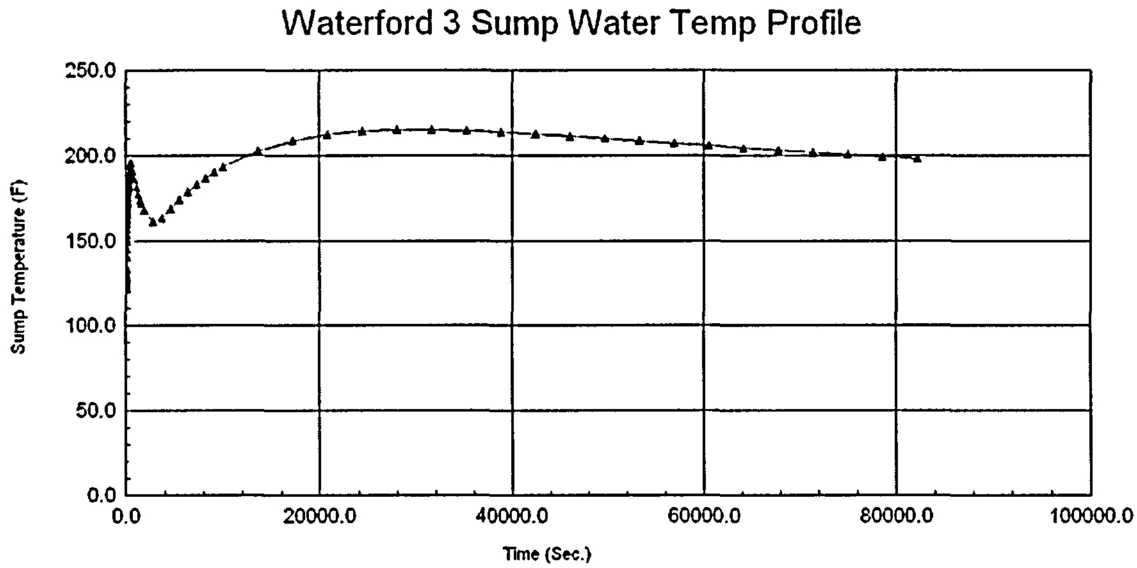
Pertaining to the flashing fraction to be assumed for LBLOCA analyses for ECCS leakage, RG 1.183 states:

"If the temperature of the leakage is less than 212 °F or the calculated flashing fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine in the leaked fluid unless a smaller amount can be justified based on actual sump pH and area ventilation rates."

The assumption pertaining to the reduced flashing fraction after 24 hours considered three criteria; ECCS fluid temperature (sump water temperature), sump pH, and ventilation in the auxiliary building areas receiving the ECCS leakage.

Figure 1 illustrates the projected sump temperature for EPU conditions for the first 24 hours. As shown in this figure, the maximum sump water temperature is 213.83 °F at 31,061 seconds. Also, the data used to generate Figure 1 states that the sump water temperature is above 212 °F for a short duration (roughly 5.28 hours to 12.3 hours). From 24 hours to approximately 11 days (1,000,000 seconds), the sump water temperature is approximated to linearly decrease to 165 °F and then remains constant at 165 °F for the duration of the accident (30 days). As shown in this figure, the maximum sump water temperature is 213.83 °F at 31,061 seconds which results in a maximum flashing fraction of ~0.2% as shown in the calculation below:

Figure 1



Flashing Fraction $= [h_f(213.83°F) - h_f(212°F)]/h_{fg}(212°F)$
$= (182.01 - 180.16)/(1150.48 - 180.16)$
$= 0.19\%$

As shown in the calculation above and Figure 1, the maximum calculated flashing fraction is 0.2% based on a "maximum" sump water temperature. After 24 hours the sump water temperature is well below 212 °F (~197 °F) and remains well below that value for the duration of the LBLOCA. Therefore, it is very conservative (by more than an order of magnitude) to model a flashing fraction of 10% for the first 24 hours of the accident and then reduce the flashing fraction to a value of 2% thereafter. As shown below, the other criteria (sump pH and area ventilation) support this conclusion as well.

An evaluation of the sump pH indicated that a sump pH 7.0 will be maintained throughout the 30-day duration of the accident, thus inhibiting further evolution of iodine out of the sump water. Therefore, sump pH supports a reduced flashing fraction after 24 hours due to pH values of 7.0 or higher for the duration of the LBLOCA.

Safety Injection and Containment Spray piping containing recirculated coolant from the containment Safety Injection sump are located in the "wing" areas of the Reactor Auxiliary Building (the –35'0" and -4'0" elevations) and the safeguard pump rooms of the Reactor Auxiliary Building (the -35'0" elevation). These areas are serviced by the CVAS ventilation. The Safety Injection equipment rooms are maintained at 104°F or below under post-LOCA conditions. The Reactor Auxiliary Building "wing" areas serviced by CVAS are maintained at temperatures of 148 °F or less during a LBLOCA. In addition, there is no portion of this area that would be subject to significant forced air flow rates that may support a significant evaporative process that would release iodine from the water pools in these areas that would be formed on these floors as a result of the ECCS leakage.

Therefore, as a result of these individual evaluations Waterford 3 concludes that the AST assessment for radioactive filter shine assuming a flashing fraction of 10% for the first 24 hours of the accident and then a reduction in the flashing fraction to a value of 2% still results in a very conservative estimate for the MCR dose assessment for radioactive filter shine.

Question 19:

Provide the isotopic source terms used as input to each of the filter shine calculations, the direct containment shine calculation and the external plume shine calculation.

Response 19:

Two sets of nuclide source files were used for the MicroShield direct containment shine calculation; 1) sprayed and unsprayed containment activity regions, and 2) annulus activity. Isotopic activity at the end of all of the time intervals for each of the regions was obtained from LBLOCA RADTRAD runs. The maximum activity in the sprayed and unsprayed regions of containment was summed to obtain the maximum total activity inside containment for each of the time intervals given in Table 1, below. The maximum activity in the annulus (Table 2) was determined and used directly in MicroShield.

TABLE 1
Maximum Containment Activity by Time Interval (Ci)

Time (hr)	2-4	4-24	24-96	96-168	168-720
Co-58	4.74E+02	1.67E+02	1.53E-01	1.65E-10	0.00E+00
Co-60	3.63E+02	1.28E+02	1.18E-01	1.31E-10	0.00E+00
Kr-85	1.33E+06	1.33E+06	1.32E+06	1.31E+06	1.30E+06
Kr-85m	2.45E+07	1.80E+07	8.12E+05	1.17E+01	1.69E-04
Kr-87	1.54E+07	5.17E+06	9.48E+01	0.00E+00	0.00E+00
Kr-88	5.34E+07	3.28E+07	2.48E+05	5.74E-03	0.00E+00
Rb-86	2.87E+03	1.01E+03	9.04E-01	8.98E-10	0.00E+00
Sr-89	4.37E+05	1.53E+05	1.40E+02	1.50E-07	0.00E+00
Sr-90	4.27E+04	1.50E+04	1.39E+01	1.54E-08	0.00E+00
Sr-91	4.58E+05	1.39E+05	3.00E+01	1.34E-10	0.00E+00
Sr-92	2.41E+05	5.11E+04	2.84E-01	0.00E+00	0.00E+00
Y-90	4.34E+02	1.49E+02	1.11E-01	5.66E-11	0.00E+00
Y-91	5.47E+03	1.92E+03	1.76E+00	1.89E-09	0.00E+00
Y-92	2.96E+03	7.05E+02	1.30E-02	0.00E+00	0.00E+00
Y-93	5.11E+03	1.57E+03	3.68E-01	0.00E+00	0.00E+00
Zr-95	7.11E+03	2.50E+03	2.29E+00	2.46E-09	0.00E+00
Zr-97	5.92E+03	1.92E+03	7.82E-01	3.49E-11	0.00E+00
Nb-95	7.10E+03	2.49E+03	2.27E+00	2.37E-09	0.00E+00
Mo-99	9.29E+04	3.20E+04	2.40E+01	1.25E-08	0.00E+00
Tc-99m	5.74E+04	1.61E+04	1.49E+00	0.00E+00	0.00E+00
Ru-103	8.37E+04	2.94E+04	2.68E+01	2.82E-08	0.00E+00
Ru-105	2.58E+04	6.67E+03	2.72E-01	0.00E+00	0.00E+00
Ru-106	3.20E+04	1.12E+04	1.04E+01	1.15E-08	0.00E+00
Rh-105	2.82E+04	9.55E+03	5.97E+00	1.62E-09	0.00E+00

TABLE 1 (Cont.)
Maximum Containment Activity by Time Interval (Ci)

Time (hr)	2-4	4-24	24-96	96-168	168-720
Sb-127	8.01E+04	2.77E+04	2.21E+01	1.43E-08	0.00E+00
Sb-129	2.03E+05	5.18E+04	1.94E+00	0.00E+00	0.00E+00
Te-127	6.19E+04	1.88E+04	3.95E+00	0.00E+00	0.00E+00
Te-127m	1.05E+04	3.69E+03	3.39E+00	3.70E-09	0.00E+00
Te-129	4.71E+04	5.07E+03	3.03E-05	0.00E+00	0.00E+00
Te-129m	5.05E+04	1.77E+04	1.61E+01	1.68E-08	0.00E+00
Te-131m	1.43E+05	4.81E+04	2.80E+01	5.90E-09	0.00E+00
Te-132	1.47E+06	5.06E+05	3.92E+02	2.30E-07	0.00E+00
I-131	7.39E+06	2.64E+06	6.05E+04	4.47E+04	3.43E+04
I-132	1.05E+07	3.72E+06	7.66E+04	3.87E+04	2.03E+04
I-133	1.35E+07	4.56E+06	5.77E+04	5.01E+03	4.51E+02
I-134	1.27E+06	9.57E+04	3.20E-04	0.00E+00	0.00E+00
I-135	1.00E+07	2.93E+06	8.85E+03	4.46E+00	2.33E-03
Xe-133	2.07E+08	2.05E+08	1.83E+08	1.22E+08	8.14E+07
Xe-135	4.91E+07	4.21E+07	9.13E+06	3.74E+04	1.53E+02
Cs-134	1.54E+06	5.45E+05	5.04E+02	5.58E-07	0.00E+00
Cs-136	4.03E+05	1.42E+05	1.25E+02	1.19E-07	0.00E+00
Cs-137	8.23E+05	2.90E+05	2.68E+02	2.98E-07	0.00E+00
Ba-139	1.40E+05	1.81E+04	7.18E-04	0.00E+00	0.00E+00
Ba-140	7.52E+05	2.63E+05	2.33E+02	2.19E-07	0.00E+00
La-140	7.43E+03	2.52E+03	1.65E+00	5.32E-10	0.00E+00
La-141	3.81E+03	9.45E+02	2.57E-02	0.00E+00	0.00E+00
La-142	1.49E+03	2.15E+02	2.47E-05	0.00E+00	0.00E+00
Ce-141	1.66E+04	5.83E+03	5.30E+00	5.52E-09	0.00E+00
Ce-143	1.59E+04	5.36E+03	3.26E+00	7.98E-10	0.00E+00
Ce-144	1.27E+04	4.47E+03	4.13E+00	4.55E-09	0.00E+00
Pr-143	6.58E+03	2.30E+03	2.04E+00	1.94E-09	0.00E+00
Nd-147	2.82E+03	9.86E+02	8.65E-01	7.95E-10	0.00E+00
Np-239	1.83E+05	6.27E+04	4.54E+01	2.09E-08	0.00E+00
Pu-238	1.08E+01	3.80E+00	3.51E-03	3.90E-12	0.00E+00
Pu-239	2.44E+00	8.56E-01	7.92E-04	8.80E-13	0.00E+00
Pu-240	3.07E+00	1.08E+00	9.99E-04	1.11E-12	0.00E+00
Pu-241	5.17E+02	1.82E+02	1.68E-01	1.87E-10	0.00E+00
Am-241	1.37E-01	4.80E-02	4.44E-05	4.94E-14	0.00E+00
Cm-242	5.23E+01	1.84E+01	1.69E-02	1.86E-11	0.00E+00
Cm-244	3.06E+00	1.08E+00	9.95E-04	1.10E-12	0.00E+00

TABLE 2
Maximum Annulus Activity by Time Interval (Ci)

Time (hr)	2-4	4-24	24-96	96-168	168-720
Co-58	3.63E-02	1.88E-02	1.89E-05	0.00E+00	0.00E+00
Co-60	2.78E-02	1.44E-02	1.46E-05	0.00E+00	0.00E+00
Kr-85	3.21E+02	2.53E+03	5.62E+03	7.40E+03	7.40E+03
Kr-85m	4.38E+03	5.59E+03	1.55E+03	5.00E-02	9.58E-07
Kr-87	1.48E+03	1.29E+03	1.82E-01	0.00E+00	0.00E+00
Kr-88	8.03E+03	8.03E+03	4.75E+02	2.46E-05	0.00E+00
Rb-86	2.39E-01	1.16E-01	1.12E-04	0.00E+00	0.00E+00
Sr-89	3.34E+01	1.73E+01	1.73E-02	8.04E-12	0.00E+00
Sr-90	3.27E+00	1.69E+00	1.72E-03	8.28E-13	0.00E+00
Sr-91	3.56E+01	1.58E+01	3.70E-03	0.00E+00	0.00E+00
Sr-92	1.96E+01	5.84E+00	3.51E-05	0.00E+00	0.00E+00
Y-90	3.33E-02	1.68E-02	1.37E-05	0.00E+00	0.00E+00
Y-91	4.19E-01	2.17E-01	2.18E-04	0.00E+00	0.00E+00
Y-92	2.37E-01	8.03E-02	1.61E-06	0.00E+00	0.00E+00
Y-93	3.97E-01	1.77E-01	4.54E-05	0.00E+00	0.00E+00
Zr-95	5.44E-01	2.82E-01	2.83E-04	0.00E+00	0.00E+00
Zr-97	4.57E-01	2.17E-01	9.66E-05	0.00E+00	0.00E+00
Nb-95	5.43E-01	2.81E-01	2.80E-04	0.00E+00	0.00E+00
Mo-99	7.13E+00	3.61E+00	2.96E-03	0.00E+00	0.00E+00
Tc-99m	4.51E+00	1.82E+00	1.84E-04	0.00E+00	0.00E+00
Ru-103	6.41E+00	3.31E+00	3.31E-03	1.52E-12	0.00E+00
Ru-105	2.05E+00	7.58E-01	3.36E-05	0.00E+00	0.00E+00
Ru-106	2.45E+00	1.27E+00	1.28E-03	6.16E-13	0.00E+00
Rh-105	2.17E+00	1.08E+00	7.38E-04	0.00E+00	0.00E+00
Sb-127	6.14E+00	3.13E+00	2.73E-03	0.00E+00	0.00E+00
Sb-129	1.61E+01	5.89E+00	2.39E-04	0.00E+00	0.00E+00
Te-127	4.82E+00	2.13E+00	4.88E-04	0.00E+00	0.00E+00
Te-127m	8.03E-01	4.16E-01	4.19E-04	0.00E+00	0.00E+00
Te-129	4.13E+00	5.88E-01	3.74E-09	0.00E+00	0.00E+00
Te-129m	3.86E+00	2.00E+00	1.99E-03	9.03E-13	0.00E+00
Te-131m	1.10E+01	5.43E+00	3.47E-03	0.00E+00	0.00E+00
Te-132	1.12E+02	5.72E+01	4.85E-02	1.24E-11	0.00E+00
I-131	5.97E+02	2.99E+02	4.77E+00	1.72E+00	1.32E+00
I-132	8.49E+02	4.21E+02	6.04E+00	1.49E+00	7.83E-01
I-133	1.10E+03	5.17E+02	4.55E+00	1.93E-01	1.74E-02
I-134	1.29E+02	1.14E+01	2.52E-08	0.00E+00	0.00E+00
I-135	8.32E+02	3.34E+02	6.98E-01	1.72E-04	8.98E-08
Xe-133	4.94E+04	3.49E+05	5.22E+05	5.22E+05	4.63E+05
Xe-135	1.02E+04	1.79E+04	1.75E+04	1.60E+02	8.71E-01

TABLE 2 (Cont.)
Maximum Annulus Activity by Time Interval (Ci)

Time (hr)	2-4	4-24	24-96	96-168	168-720
Cs-134	1.29E+02	6.26E+01	6.22E-02	3.00E-11	0.00E+00
Cs-136	3.36E+01	1.63E+01	1.55E-02	6.38E-12	0.00E+00
Cs-137	6.86E+01	3.33E+01	3.32E-02	1.60E-11	0.00E+00
Ba-139	1.20E+01	2.09E+00	8.88E-08	0.00E+00	0.00E+00
Ba-140	5.76E+01	2.97E+01	2.87E-02	1.18E-11	0.00E+00
La-140	5.71E-01	2.85E-01	2.04E-04	0.00E+00	0.00E+00
La-141	3.04E-01	1.07E-01	3.17E-06	0.00E+00	0.00E+00
La-142	1.26E-01	2.47E-02	3.05E-09	0.00E+00	0.00E+00
Ce-141	1.27E+00	6.57E-01	6.55E-04	0.00E+00	0.00E+00
Ce-143	1.22E+00	6.06E-01	4.03E-04	0.00E+00	0.00E+00
Ce-144	9.74E-01	5.05E-01	5.10E-04	2.45E-13	0.00E+00
Pr-143	5.04E-01	2.60E-01	2.52E-04	0.00E+00	0.00E+00
Nd-147	2.16E-01	1.11E-01	1.07E-04	0.00E+00	0.00E+00
Np-239	1.40E+01	7.08E+00	5.61E-03	0.00E+00	0.00E+00
Pu-238	8.26E-04	4.28E-04	4.34E-07	0.00E+00	0.00E+00
Pu-239	1.86E-04	9.66E-05	9.79E-08	4.73E-17	0.00E+00
Pu-240	2.35E-04	1.22E-04	1.23E-07	5.96E-17	0.00E+00
Pu-241	3.96E-02	2.05E-02	2.08E-05	1.00E-14	0.00E+00
Am-241	1.05E-05	5.42E-06	5.49E-09	0.00E+00	0.00E+00
Cm-242	4.00E-03	2.07E-03	2.09E-06	0.00E+00	0.00E+00
Cm-244	2.34E-04	1.21E-04	1.23E-07	0.00E+00	0.00E+00

For the external plume shine calculation, isotopic activity in the environment at the end of all of the time intervals was obtained from LBLOCA RADTRAD runs also. The maximum total activity in the environment for each of the time intervals used is given in Table 3, below. The concentration in the environment was assumed to be representative of the cloud concentration above the MCR.

The cloud concentration was calculated as follows:

$$C_i = Q_i * \frac{1}{\Delta t_j} * \frac{X}{Q_j} * K$$

where:

C_i = concentration of isotope i during time interval j (μCi/cc)

Q_i = activity of isotope i released during time interval j (Ci)

Δt_j = length of time interval (hr)

X/Q_i = meteorological dispersion factor for time interval j (sec/m^3)

K = unit conversion factor

$= (10^6\, \mu Ci/Ci) * (1hr/3600\sec) * (1m^3/10^6 cm^3)$

The external plume concentration is shown in Table 3.

TABLE 3
External Plume Concentration (µCi/cc)

Time (hr)	0-0.5	0.5-2	2-4	4-8	8-24	24-96	96-720
Co-58	0.00E+00	5.53E-09	1.65E-09	5.89E-10	1.84E-11	8.13E-15	0.00E+00
Co-60	0.00E+00	4.24E-09	1.26E-09	4.51E-10	1.41E-11	6.10E-15	0.00E+00
Kr-85	3.21E-06	7.53E-05	8.22E-05	8.21E-05	3.32E-05	1.38E-05	1.59E-05
Kr-85m	8.78E-05	1.54E-03	1.29E-03	8.22E-04	8.09E-05	4.86E-07	0.00E+00
Kr-87	1.46E-04	1.30E-03	5.64E-04	1.26E-04	1.57E-06	2.03E-10	0.00E+00
Kr-88	2.40E-04	3.58E-03	2.58E-03	1.28E-03	6.98E-05	7.73E-08	0.00E+00
Rb-86	9.95E-09	3.74E-08	9.97E-09	3.55E-09	1.10E-10	6.10E-14	0.00E+00
Sr-89	0.00E+00	5.10E-06	1.52E-06	5.42E-07	1.69E-08	6.10E-12	0.00E+00
Sr-90	0.00E+00	4.98E-07	1.48E-07	5.30E-08	1.66E-09	8.13E-13	0.00E+00
Sr-91	0.00E+00	5.71E-06	1.49E-06	4.40E-07	9.33E-09	2.03E-12	0.00E+00
Sr-92	0.00E+00	3.58E-06	6.74E-07	1.25E-07	1.14E-09	0.00E+00	0.00E+00
Y-90	0.00E+00	5.11E-09	1.49E-09	5.19E-10	1.53E-11	6.10E-15	0.00E+00
Y-91	0.00E+00	6.39E-08	1.90E-08	6.79E-09	2.12E-10	8.13E-14	0.00E+00
Y-92	0.00E+00	4.14E-08	8.68E-09	1.87E-09	2.22E-11	0.00E+00	0.00E+00
Y-93	0.00E+00	6.35E-08	1.67E-08	4.99E-09	1.08E-10	2.03E-14	0.00E+00
Zr-95	0.00E+00	8.30E-08	2.47E-08	8.82E-09	2.75E-10	1.22E-13	0.00E+00
Zr-97	0.00E+00	7.17E-08	1.98E-08	6.36E-09	1.59E-10	2.03E-14	0.00E+00
Nb-95	0.00E+00	8.29E-08	2.47E-08	8.80E-09	2.74E-10	1.22E-13	0.00E+00
Mo-99	0.00E+00	1.09E-06	3.20E-07	1.11E-07	3.28E-09	1.02E-12	0.00E+00
Tc-99m	0.00E+00	7.44E-07	1.80E-07	4.77E-08	8.21E-10	2.03E-13	0.00E+00
Ru-103	0.00E+00	9.77E-07	2.91E-07	1.04E-07	3.23E-09	1.22E-12	0.00E+00
Ru-105	0.00E+00	3.48E-07	7.84E-08	1.87E-08	2.66E-10	0.00E+00	0.00E+00
Ru-106	0.00E+00	3.73E-07	1.11E-07	3.97E-08	1.24E-09	4.07E-13	0.00E+00
Rh-105	0.00E+00	3.35E-07	9.65E-08	3.27E-08	9.18E-10	4.07E-13	0.00E+00

TABLE 3 (Cont.)
External Plume Concentration (µCi/cc)

Time (hr)	0-0.5	0.5-2	2-4	4-8	8-24	24-96	96-720
Sb-127	0.00E+00	9.41E-07	2.77E-07	9.69E-08	2.91E-09	1.02E-12	0.00E+00
Sb-129	0.00E+00	2.74E-06	6.12E-07	1.44E-07	2.01E-09	0.00E+00	0.00E+00
Te-127	0.00E+00	7.73E-07	2.02E-07	5.93E-08	1.25E-09	2.03E-13	0.00E+00
Te-127m	0.00E+00	1.22E-07	3.65E-08	1.30E-08	4.07E-10	1.83E-13	0.00E+00
Te-129	0.00E+00	9.98E-07	1.01E-07	8.42E-09	1.88E-11	0.00E+00	0.00E+00
Te-129m	0.00E+00	5.89E-07	1.75E-07	6.25E-08	1.95E-09	8.13E-13	0.00E+00
Te-131m	0.00E+00	1.71E-06	4.88E-07	1.64E-07	4.52E-09	1.22E-12	0.00E+00
Te-132	0.00E+00	1.72E-05	5.06E-06	1.77E-06	5.25E-08	1.83E-11	0.00E+00
I-131	1.85E-05	9.28E-05	2.60E-05	9.47E-06	4.40E-07	5.09E-08	1.94E-08
I-132	2.66E-05	1.33E-04	3.68E-05	1.32E-05	5.89E-07	5.35E-08	6.89E-09
I-133	3.68E-05	1.75E-04	4.65E-05	1.56E-05	5.81E-07	1.97E-08	1.82E-10
I-134	2.84E-05	3.97E-05	2.44E-06	1.34E-07	1.25E-10	0.00E+00	0.00E+00
I-135	3.33E-05	1.40E-04	3.23E-05	9.00E-06	2.11E-07	9.15E-10	0.00E+00
Xe-133	5.08E-04	1.18E-02	1.27E-02	1.25E-02	4.77E-03	1.55E-03	4.48E-04
Xe-135	1.44E-04	2.92E-03	2.80E-03	2.23E-03	4.18E-04	1.35E-05	6.49E-09
Cs-134	5.34E-06	2.01E-05	5.38E-06	1.93E-06	6.02E-08	2.44E-11	0.00E+00
Cs-136	1.40E-06	5.26E-06	1.40E-06	4.99E-07	1.54E-08	6.10E-12	0.00E+00
Cs-137	2.84E-06	1.07E-05	2.87E-06	1.03E-06	3.21E-08	1.22E-11	0.00E+00
Ba-139	0.00E+00	2.67E-06	3.22E-07	3.32E-08	1.07E-10	0.00E+00	0.00E+00
Ba-140	0.00E+00	8.79E-06	2.61E-06	9.26E-07	2.86E-08	1.02E-11	0.00E+00
La-140	0.00E+00	8.80E-08	2.54E-08	8.68E-09	2.47E-10	8.13E-14	0.00E+00
La-141	0.00E+00	5.24E-08	1.14E-08	2.57E-09	3.33E-11	0.00E+00	0.00E+00
La-142	0.00E+00	2.69E-08	3.57E-09	4.16E-10	1.67E-12	0.00E+00	0.00E+00
Ce-141	0.00E+00	1.94E-07	5.77E-08	2.06E-08	6.40E-10	2.64E-13	0.00E+00
Ce-143	0.00E+00	1.89E-07	5.42E-08	1.83E-08	5.11E-10	1.42E-13	0.00E+00
Ce-144	0.00E+00	1.48E-07	4.42E-08	1.58E-08	4.94E-10	2.03E-13	0.00E+00
Pr-143	0.00E+00	7.69E-08	2.28E-08	8.11E-09	2.51E-10	1.02E-13	0.00E+00
Nd-147	0.00E+00	3.30E-08	9.78E-09	3.47E-09	1.07E-10	4.07E-14	0.00E+00
Np-239	0.00E+00	2.16E-06	6.29E-07	2.18E-07	6.35E-09	2.24E-12	0.00E+00
Pu-238	0.00E+00	1.26E-10	3.75E-11	1.34E-11	4.20E-13	1.63E-16	0.00E+00
Pu-239	0.00E+00	2.84E-11	8.46E-12	3.03E-12	9.48E-14	4.07E-17	0.00E+00
Pu-240	0.00E+00	3.58E-11	1.07E-11	3.82E-12	1.19E-13	6.10E-17	0.00E+00
Pu-241	0.00E+00	6.03E-09	1.80E-09	6.43E-10	2.01E-11	8.13E-15	0.00E+00
Am-241	0.00E+00	1.59E-12	4.75E-13	1.70E-13	5.31E-15	2.24E-18	0.00E+00
Cm-242	0.00E+00	6.10E-10	1.82E-10	6.49E-11	2.03E-12	8.13E-16	0.00E+00
Cm-244	0.00E+00	3.57E-11	1.06E-11	3.80E-12	1.19E-13	6.10E-17	0.00E+00

A total of four sets of fission product data were derived from RADTRAD to perform the filter shine calculations. Table 4 illustrates the iodine radioisotopes that were deposited on the Control Room Recirculation filters due to containment air leakage. Table 5 illustrates the iodine radioisotopes that were deposited on the SBVS filters due to containment air leakage. Table 6 illustrates the iodine radioisotopes that were deposited on the CVAS filters due to containment air leakage. Table 7 illustrates the iodine radioisotopes that were deposited on the CVAS filters due to ESF leakage.

Table 4
Summary of Radioactive Materials Deposited on Waterford 3 Control Room Emergency Ventilation Filters from Normal Containment Air Leakage

Radioisotope	2 hours (Ci)	4 hours (Ci)	8 hours (Ci)	1 Day (Ci)	4 Days (Ci)	7 Days (Ci)	30 Days (Ci)
I-131	0.106	0.141	0.163	0.158	0.124	0.097	0.016
I-132	0.151	0.198	0.224	0.200	0.108	0.058	5.1E-4
I-133	0.197	0.245	0.251	0.151	0.014	1.3E-3	N/A
I-134	0.026	6.6E-3	3.2E-4	1.03E-9	N/A	N/A	N/A
I-135	0.150	0.161	0.123	0.024	1.3E-5	N/A	N/A

Table 5
Summary of Radioactive Materials Deposited on Waterford 3 Shield Building Ventilation System Filters from Normal Containment Air Leakage

Radioisotope	2 hours (Ci)	4 hours (Ci)	8 hours (Ci)	1 Day (Ci)	4 Days (Ci)	7 Days (Ci)	30 Days (Ci)
I-131	610.74	1517.6	2093.9	2195.6	1831.7	1517.4	314.32
I-132	870.73	2138.1	2887.2	2782.2	1587.9	899.92	10.152
I-133	1137.9	2644.2	3231.5	2104.4	206.31	20.087	3.1E-7
I-134	165.21	70.85	3.940	1.38E-5	N/A	N/A	N/A
I-135	877.12	1735.2	1583.7	327.78	0.186	1.05E-4	N/A

Table 6
Summary of Radioactive Materials Deposited on Waterford 3 Controlled Ventilation Area System Filters from Normal Containment Air Leakage

Radioisotope	2 hours (Ci)	4 hours (Ci)	8 hours (Ci)	1 Day (Ci)	4 Days (Ci)	7 Days (Ci)	30 Days (Ci)
I-131	807.04	2158.8	3110.0	3291.5	2746.7	2275.4	471.35
I-132	1150.7	3041.6	4288.3	4170.9	2381.1	1349.5	15.225
I-133	1504.1	3762.4	4800.0	3154.8	309.37	30.121	4.64E-7
I-134	220.13	101.43	5.8620	2.07E-5	N/A	N/A	N/A
I-135	1160.3	2470.4	2352.7	491.40	0.2789	1.57E-4	N/A

Table 7
Summary of Radioactive Materials Deposited on Waterford 3 Control Room Emergency Ventilation Filters due to ECCS Leakage (10% Flashing Fraction, 0-24 hours, then 2% for 1-30 days)

Radioisotope	2 hours (Ci)	4 hours (Ci)	8 hours (Ci)	1 Day (Ci)	4 Days (Ci)	7 Days (Ci)	30 Days (Ci)
I-131	106.14	575.35	1696.3	5931.2	7757.3	8307.7	3580.1
I-132	151.93	814.91	2352.4	7560.1	6764.9	4956.7	116.37
I-133	202.23	1033.0	2704.0	5875.4	903.46	113.76	3.65E-06
I-135	N/A	727.89	1431.2	990.01	0.8821	6.43E-04	N/A

Question 20:

Was cesium included in the filter shine source terms? If not, why wasn't shine dose from deposition of radioactive cesium in the filters considered?

Response 20:

The results presented in the September 1, 2004, submittal did not include radioactive cesium as a filter shine source term. The only radioactive material considered in the MicroShield dose calculations was the radioactive iodines. The filter shine shielding calculations considered fission product deposition on the filters via two mechanisms: containment leakage and ECCS leakage. Per RG 1.183, the ECCS leakage RADTRAD model only considered the radioactive iodines in the sump water. Note that ECCS leakage is the dominant source of radioactive iodine on the Controlled Ventilation filters as shown in Tables 4-7 in the response to Question #19.

The containment leakage RADTRAD model considered the cesium radioisotopes, however, sensitivity runs using MicroShield concluded that the radioactive iodines were deposited on the various filter systems in the largest quantities (in comparison to radioactive cesium). In addition, a separate MicroShield calculation considered radioactive cesium, along with other radioactive materials and concluded due to the ample amount of shielding between the source and dose points that the radioactive iodines contributed more than 98% of the total MCR dose due to radioactive shine from the various filter systems analyzed. Therefore, it was concluded that the current radioactive filter shine calculation performed in support of the Waterford 3 AST for EPU provided enough conservatism to offset the consequences of neglecting radioactive cesium in the current analyses.

Question 21:

Provide assumptions and inputs for each of the shine dose analyses performed in MicroShield. This should include, but is not limited to, assumptions on shielding, geometry, source type and location and receptor location, along with their bases. Provide plant plans that identify the assumed shine sources and control room receptor point locations for the 3 filter shine dose analyses (shield building ventilation system, controlled ventilation areas system, and control room emergency air recirculation system), the direct containment shine dose analysis and the external plume shine dose analysis.

Response 21:

Containment Shine MicroShield Model:

The MicroShield model for containment shine was performed assuming two separate sources: (1) containment and (2) Shield Building. The containment was modelled as a right circular cylinder with side shields.

The containment geometry parameters are given below:

- Radius of Containment inside steel shield: = 69.83 ft
- Volume: 2.568E+6 ft^3 (2.055E+6 ft^3 (sprayed region) & 5.137E+5 ft^3 (unsprayed region)
- Shields (material & thickness):

1.	steel	2 in
2.	air gap	4 ft
3.	concrete	3 ft (Shield Bldg wall)
4.	air gap	19 ft (Shield Bldg wall to column L)
5.	concrete	3 ft (RAB wall at column L)

The Shield Building Annulus is modelled as an annular cylinder with an external dose point. The annulus geometry parameters are given below:

- Inside radius = 70 ft
- Width = 4 ft
- Outside radius = 74 ft
- Volume = 550,000 ft^3

Note: Concrete is specified as the reference material for buildup for all cases.

External Cloud MicroShield Model:

The external cloud is modelled as an infinite slab source outside the MCR. The roof of the MCR is represented as a concrete shield 2'-3" thick. This shield is used as the reference buildup material.

The dose point is conservatively located slightly more than 1 ft from the bottom of the roof slab. Since the top of the roof slab is El. 69' and the floor is at El. 46', the dose point is conservatively more than 19' above the floor of the MCR.

Filter Shine MicroShield Models:

The main control room filter shine dose calculations were performed on the following three filters systems:

1. Control Room Emergency Ventilation System
2. Shield Building Ventilation System (SBVS)
3. Controlled Ventilation Area System (CVAS)

The MicroShield analyses credit a 1/4" thick steel thickness for all three filter housings. Table 1 below illustrates the charcoal filter design characteristics for each filter system that is assumed to be the source for the MicroShield calculations. Control room occupancy factors from RG 1.183 are used to calculate the integrated dose in the control room from the filter shine calculations. Figure 1 illustrates the dose geometry for the Control Room Emergency Ventilation filters. Figure 2 illustrates the dose geometry for the SBVS filters. Figure 3 illustrates the dose geometry for the CVAS filters. Tables 4-7 in response to Question #19 represent the source terms deposited on the various filter systems discussed above.

As shown in Figures 1-3 below, the doses were reported for dose points located in the MCR near the operations panels where the operators would actually be stationed, not just inside the walls defining the control room envelope (i.e., including the corridor surrounding the MCR).

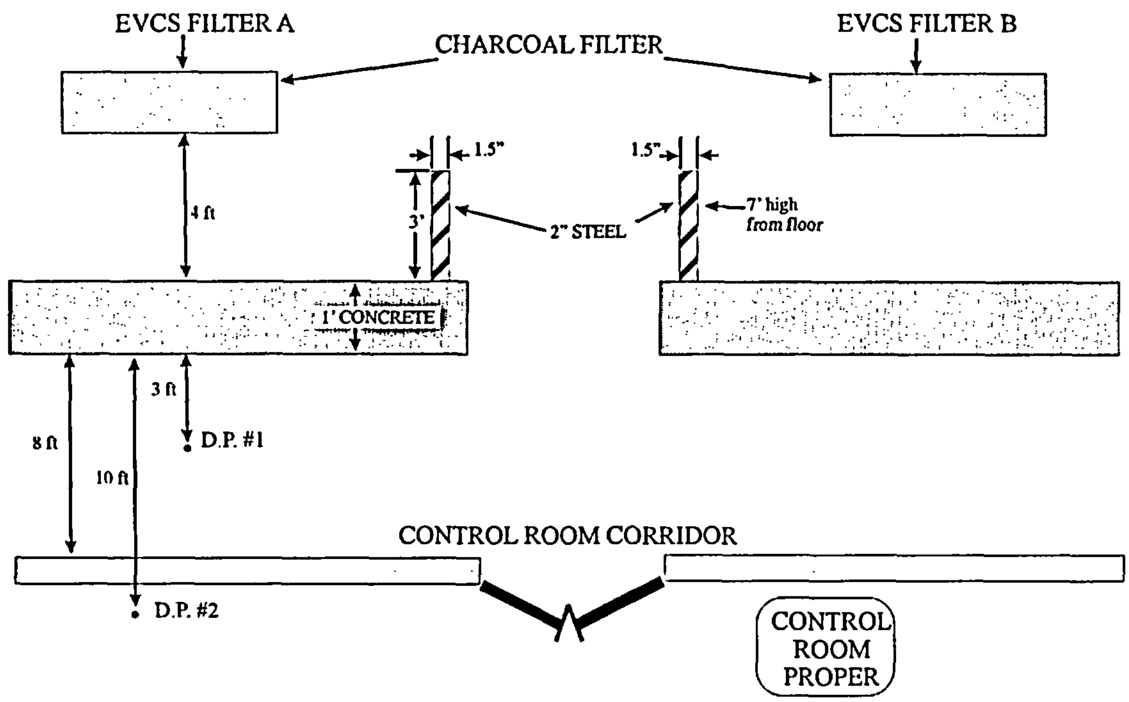


Figure 1 Control Room Emergency Ventilation Unit MicroShield Shielding Model

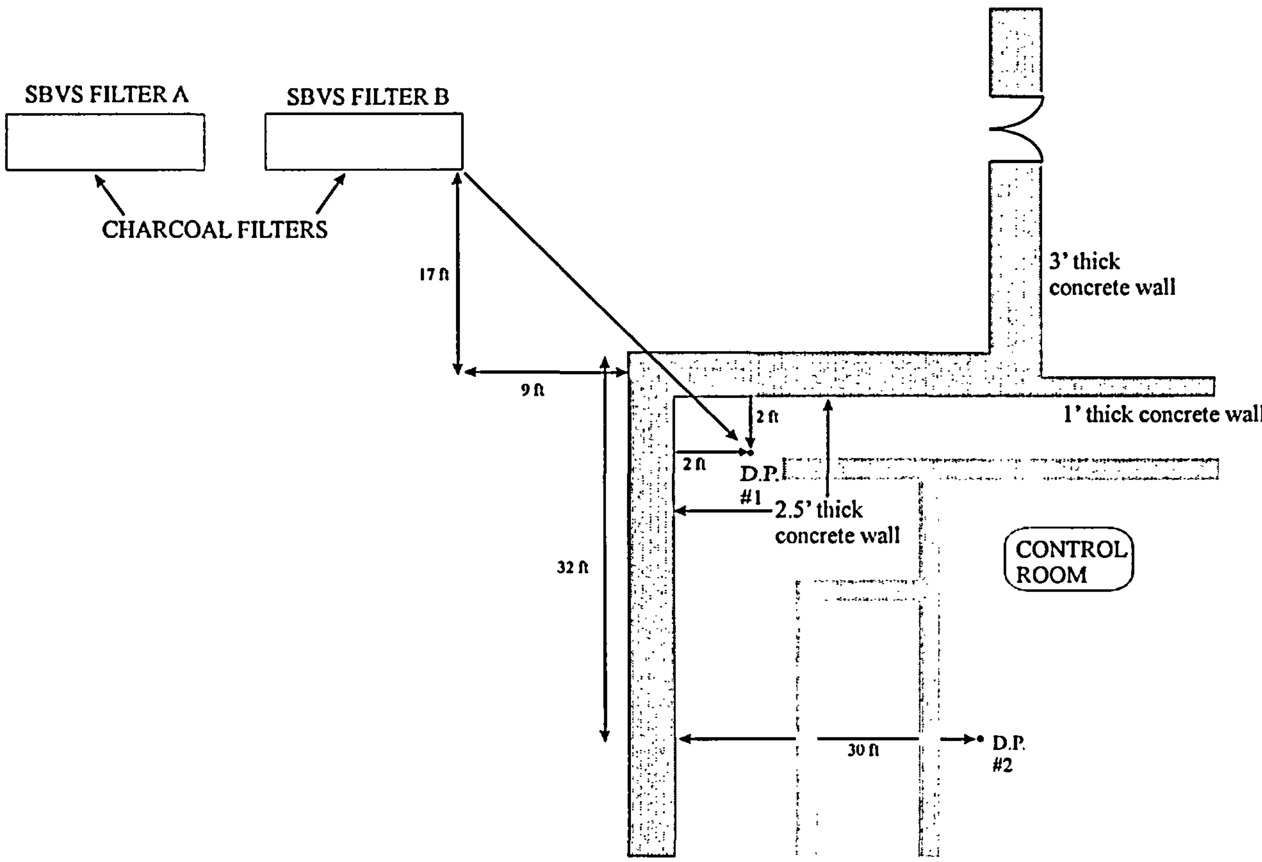


Figure 2 SBVS MicroShield Shielding Model

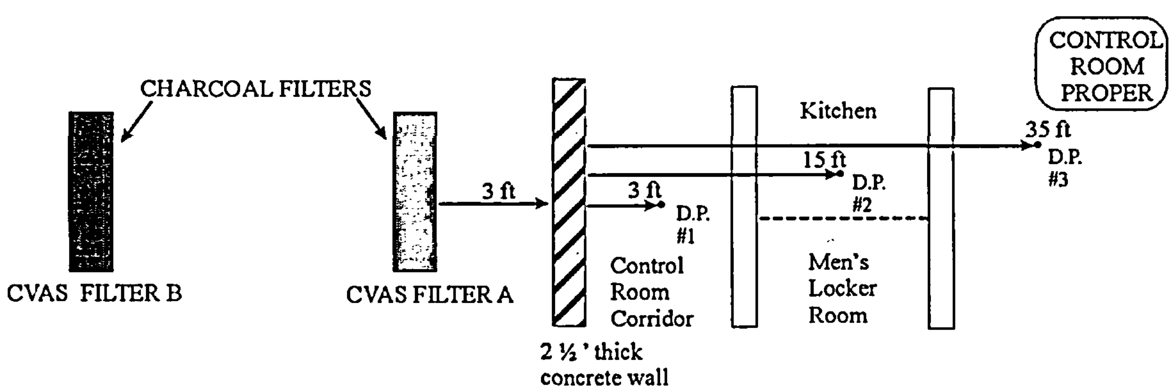


Figure 3 CVAS MicroShield Shielding Model

Table 1: Charcoal Filter Design Information

Control Room Emergency Air Filtration Charcoal Filter Geometry and System Characteristics:

- Thickness: 12" (3 – 4" beds)
- Width: 60"
- Height: 6.5'
- Filtration efficiency – 100% (Aerosol/Elemental/Organic Iodine)
- Flow: 3,800 CFM (4250 CFM used to address pressurization flow and unfiltered in-leakage)

Shield Building Ventilation System Charcoal Filter Geometry and System Characteristics:

- Thickness: 1.667' (5-4" beds)
- Width: 60"
- Height: 7.5'
- Filtration efficiency – 100% (Aerosol/Elemental/Organic Iodine)
- Flow: 11,000 CFM (10,000 CFM +/- 1000 CFM)

Controlled Ventilation Area System Charcoal Filter Geometry and System Characteristics:

- Thickness: 8" (2-4" beds)
- Width: 60"
- Height: 7.5'
- Filtration efficiency – 100% (Aerosol/Elemental/Organic Iodine)
- Flow: 4,000 CFM assumed (Design capacity – 3000 CFM +/- 10%)

Question 22:

In order to complete its evaluation, the staff needs to review the general assumptions and calculations used by the licensee to demonstrate that the containment sump pH will be maintained above 7 throughout the duration of the accident. Please describe the procedure utilized for calculating pH of the containment sump water during the 30 day period after a LOCA. Please provide the inputs to the STARpH 1.04 code and the results calculated by it.

Response 22:

Response: The methodology of STARpH is consistent with NUREG/CR-2367, SAND81-7159 (August 1981), NUREG-1081 (September 1984), NUREG/CR-5808, ORNL/TM-11970 (December 1991), NUREG/CR-5732, ORNL/TM-11861 (April 1992), NUREG/CR-5950, ORNL/TM-12242 (December, 1992), NUREG-1465 (February 1995), and RG 1.183 (July 2000).

STARpH performs the pH calculation in three steps. First, the generation of HNO_3 is calculated; second, the generation of HCl is calculated; and third, the time-dependent pH is calculated based on the presence of these strong acids, boric acid, and TSP. Each of the steps is covered in more detail (along with input values) below:

<u>HNO_3 Production</u>

The inputs for the STARpH water radiolysis calculation are as follows:

- Thermal power = 3716 MWt
- Sump/RCS liquid volumes/masses: RWSP = 576,859.1 gallons @ 50°F, SIT = 48,100 gallons @ 90°F, BAMT = 22,920 gallons @49°F, RCS = 498,000 lbm
- Maximum boron concentrations: RWSP = 3,000 ppm, SIT = 3,000 ppm, BAMT = 6,187 ppm, RCS = 2,500 ppm, on a mass basis
- Mass of TSP in containment = 20,521 lbm
- Final pool volume = 2.75E6 Liters with 30-day average temperature of 77.7 C
- Pool pH prior to strong acid addition = 7.1
- Fission product group mass inventory in core (kg):
    - Iodine = 32.4
    - Cesium = 414
    - Tellurium = 71.4
    - Strontium = 127
    - Barium = 186
    - Ruthenium = 1023
    - Cerium = 1390
    - Lanthanum = 1493
- G for nitric acid production in water = 0.007 molecules/100 eV

The combined gamma and beta power absorbed in the Waterford 3 sump water, by STARpH interval, is as follows:

Table 1

Interval (hours)	MeV Absorbed
0 - 1	2.49E+23
1 - 2	9.15E+22
2 - 5	1.88E+23
5 - 12	3.08E+23
12 - 24	4.09E+23
24 - 72	1.16E+24
72 - 240	2.15E+24
240 - 480	1.39E+24
480 - 720	9.15E+23
Total	6.86E+24

Based on 6.86E+24 MeV being absorbed and 7E-03 molecules of HNO_3 being produced per 100 eV, approximately 4.8E+26 molecules or 7.98E+02 g mol of HNO_3 are produced over 720 hours.

<u>HCl Production</u>

The inputs for the STARpH cable calculation are as follows:

- Cable OD = 0.61in • 2.54 cm/in = 1.55 cm
- Hypalon jacket thickness = 0.05 in • 2.54 cm/in = 0.127 cm
- Hypalon jacket ID = 1.55 cm – 2 • 0.127 cm = 1.30 cm
- Hypalon jacket mass = 17,649.7 lbm
- G for Hypalon = 3.512E-20 g mol/MeV = 1E-9 g mol/lbm-Rad

The actual integrated doses to cable over the nine STARpH calculation intervals for Waterford 3 are as follows:

Table 2

Interval (hours)	Gamma Dose (Rads)	Beta Dose (Rads)
0 - 1	3.67E+05	3.75E+06
1 - 2	3.28E+05	3.35E+06
2 - 5	8.25E+05	8.03E+06
5 - 12	1.38E+06	1.34E+07
12 - 24	1.76E+06	1.70E+07
24 - 72	4.59E+06	4.67E+07
72 - 240	5.30E+06	6.72E+07
240 - 480	1.41E+06	2.28E+07
480 - 720	5.05E+05	8.84E+06
Total	1.65E+07	1.91E+08

These doses take into account the fact that for the beta dose, the beta power is attenuated in only 15% of the Hypalon mass because of the effect of self-shielding, while for the gamma dose, 100% of the Hypalon mass is irradiated. Therefore, for the same total energy absorbed, the beta Rads would be expected to be about a factor of six greater than the gamma Rads.

If one multiplies the total Hypalon absorbed dose in Rads from Table 2 (gamma = 1.65E+07 Rads and beta = 1.91E+08 Rads) by the Hypalon mass (17,649.7 lbm for the gamma and 15% of that value for the beta) and then by the G value of 1E-9 g mol/lbm-Rad, one obtains 7.97E+02 g mol of HCl, the value also obtained in the STARpH calculation. The other inputs (cable OD, jacket thickness, and jacket ID) have minimal effect on the gamma radiation calculation, the only effect being self-shielding which for the gamma radiation is negligible. For the beta radiation calculation, the cable OD, jacket thickness, and the jacket ID have an impact because the exposed surface area of the jacket is a determinant of the fraction of the beta power in the containment atmosphere that is absorbed by the Hypalon material.

pH Calculation Knowing Strong-Acid Concentration

Once the concentration of HNO_3 and HCl are known from the previous two steps, the pH as a function of time is determined from the concentration of strong acids HNO_3, HCl, and HI (quantified by the assumption that 5% of the fission-product iodine mass appears as HI) and the buffers boric acid and TSP. CsOH (from fission-product cesium) is considered only to the extent of neutralizing the small HI contribution. Otherwise, it is ignored.

The results of the pH calculation are that a containment sump pH of 7.0 or greater is maintained for Waterford 3 for at least 30 days after the accident. This is shown in the following table of time-dependent pH values:

Table 3

Time	pH
1 h	7.1
2 h	7.1
5 h	7.1
12 h	7.1
1 d	7.1
3 d	7.1
10 d	7.1
20 d	7.0
30 d	7.0

Question 23:

SRP 6.5.2, "Containment Spray as a Fission Product Cleanup System", states that the removal of iodine from the containment is achieved during injection and recirculation phases. In your submittal, only the elemental and particulate iodine removal coefficients are discussed. The staff assumes these are the coefficients for the injection phase. Please discuss the technical basis for only calculating the elemental and particulate iodine removal coefficients during injection.

Response 23:

The spray removal coefficient for elemental iodine is discussed in the response to Question 24 below. Because the calculated coefficient exceeded the 20/hour maximum value specified in SRP 6.5.2, the value of 20/hour has been assumed for both injection and recirculation phases until a maximum decontamination factor has been achieved. As discussed in Section 4.0 of the W3F1-2004-0053 Licensing Report, it has been determined that containment sump pH will be 7.0 or greater for the 30 day duration of the analysis, thus preventing re-evolution of elemental iodine dissolved in containment sump water.

As discussed in the response to Question 24 below, Waterford 3 will revise the AST LBLOCA calculation to not credit fission product cleanup of Elemental Iodine due to Containment Spray for calculation of offsite and control room dose due to the containment air release pathway.

Consistent with the Waterford 3 containment spray system design, the same spray flow rate is assumed for both injection and recirculation mode.

Determination of the particulate removal coefficient is consistent with SRP 6.5.2, which does not differentiate between injection and recirculation phase in its guidance regarding spray removal coefficients for particulate iodine. Spray removal of organic iodine is not modelled.

Question 24:

The SRP states that the maximum value of decontamination factor (DF) for elemental iodine should not exceed 200. However, if the calculated value of DF is less than 200, this value should be used in the analysis. In order for the staff to verify that the calculated value of DF is not less than 200, please provide the method used in calculation and the values of the corresponding input parameters.

Response 24:

Waterford 3 had originally proposed use of a maximum DF of 200 for modeling the spray removal of elemental iodine. Waterford 3 had interpreted NUREG-0800 as endorsing this value for all applications.

Note per RG 1.183 the elemental iodine is 4.85% of the iodine release for the LBLOCA radiological model using AST. Thus, there is only a small sensitivity of the results for LBLOCA dose to assumptions for fission product scrubbing of elemental iodine.

In the analyses submitted via W3F1-2004-0053 and W3F1-2004-0073, the results indicated that a total DF of 200 for elemental iodine was achieved between 1.8 hours and 2.0 hours into the analyzed event. The detailed model secured spray scrubbing of elemental iodine at 1.8 hours, at an actual maximum DF of 92.4. The DF was defined as the inverse of the fraction of the mass remaining as a function of time:

$$m_f = M(t)/M(0) = 1 / DF$$

It is noted that there is a phased release associated with the AST, such that the predicted iodine inventory in containment peaks at 1.8 hours into the event. The DF has been conservatively based upon the total iodine inventory which is available for release rather than the maximum inventory achieved during the event, which is lower in magnitude.

Waterford 3 had interpreted NUREG-0800 as endorsing the use of a value of 200 for all applications as a maximum DF for elemental iodine. SRP Section 6.5.2 provides the following equation for determining the maximum DF for the containment atmosphere which can be achieved by the containment spray system:

$$DF = 1 + V_{sump} * H / V_{containment\ vapor}$$

For Waterford 3 dose analyses, the containment volume assumption is 2,677,000 ft^3 and the minimum assumed sump volume is 57,900 ft^3. The corresponding containment free vapor volume is 2,677,000 - 57,900 ft^3 = 2,619,100 ft^3. Based upon a pH of 7, a partition coefficient of 10,000 is assumed, based upon NUREG/CR-2900, Figure 33. This assumption has been applied for other PWRs (e.g., Point Beach, as cited in July 9, 1997, Safety Evaluation Report). This results in a DF of:

$$DF = 1 + (57{,}900\ ft^3) * 10{,}000 / 2{,}619{,}000\ ft^3 = 222$$

This supports the value of 200.

Figure 33 from NUREG/CR-2900 assumes a 100°C sump liquid temperature. Waterford 3 post-LOCA sump temperatures have a 77.7°C 30 day average temperature, and the sump temperature for early in the event could be characterized as approximately 140°F (60°C). Assuming a more conservative sump temperature of 50°C, semi-log interpolation between Figure 34 and Figure 31 of NUREG/CR-2900 would give a value of H of approximately 1250. Thus, a more exact plant-specific value would be:

$$DF = 1 + (57{,}900\ ft^3) * 1250 / 2{,}619{,}000\ ft^3 = 28.6$$

The impact of using this more exact DF value upon the AST LBLOCA radiological analyses would be very small.

Thus, while the Waterford 3 analyses should have assumed a smaller value for maximum elemental iodine DF, the impact of this assumption is negligible and the results of the analyses are sufficient to demonstrate that Waterford 3 will meet 10CFR50.67 acceptance criteria.

Waterford 3 will revise the calculation of AST LBLOCA offsite and control room dose due to the containment air release pathway to remove the credit for the scrubbing of elemental iodine by containment spray. As stated above, this is expected to have minimal (on order of 0.05 Rem TEDE to MCR) impact on results. Waterford 3 will submit the revised AST LBLOCA results to the NRC by October 22, 2004. Because doses associated with ESF leakage are unaffected by assumptions related to containment spray scrubbing of iodine and that is the major contribution to control room shine dose, there is negligible impact of the DF assumption upon containment shine doses. Therefore, Waterford 3 does not intend to change the basis for the LBLOCA containment shine dose calculations at this time.

Waterford 3 evaluated the removal rate of elemental iodine due to containment spray using the model provided in NUREG-0800 Section 6.5.2:

$\lambda_{e,s} = 6\ K_g\ T\ F / V\ D$

where T is the time of fall of the drops (18.3 seconds) and K_g is the gas-phase mass-transfer coefficient (14.8 ft/s). F is the minimum containment spray flow rate of 1750 gpm (= 3.899 ft^3/sec); consistent with the Waterford 3 containment spray system design, the same spray flow rate is assumed for both injection and recirculation mode. V is the containment volume (2.677E+06 ft^3) and D is the diameter of the spray drops (700 microns). This results in a calculated value of $\lambda_{e,s}$ of approximately 3700/hour. Since this greatly exceeds the 20/hour value specified in SRP Section 6.5.2, the 20/hour value is utilized.

Question 25:

In order to complete its evaluation, the staff needs to review the calculation of the natural deposition removal coefficient of elemental iodine during injection. Please provide the input parameters used to calculate the natural deposition removal coefficient for elemental iodine.

Response 25:

Waterford 3 models natural deposition of elemental iodine per SRP, Section 6.5.2, where the coefficient is given by:

$$\lambda_w = K_w A / V$$

A containment volume, V, of 2.677E+06 ft^3 is assumed. The surface area available, A, is conservatively estimated as that corresponding to the cylindrical structure of the containment, with a 70 foot radius and a 150 foot height, for an area of about 66,000 ft^2. A K_w value of 4.9 meters/hour is assumed. This results in a λ_w for elemental iodine of 0.40/hour.

Attachment 2

W3F1-2004-0095

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Response 10: Interpretation of RG 1.183 requirements on releases for SGTR was discussed at the August 12, 2004, meeting between Entergy and NRC. As a result of these discussions, Entergy agrees to revise the [SGTR] analysis as requested. **Response 11:** The weighting values will change as a result of the ongoing [SGTR] reanalysis (see Response 10) to fully account for early releases from the affected SG using both ADVs for the early rapid cooldown prior to isolation. **Response 12:** This information will be provided in conjunction with the information requested in Question 10.	X		10/22/04
Response 23: As discussed in the response to Question 24 below, Waterford 3 will revise the AST LBLOCA calculation to not credit fission product cleanup of Elemental Iodine due to Containment Spray for calculation of offsite and control room dose due to the containment air release pathway. **Response 24:** Waterford 3 will revise the calculation of AST LBLOCA offsite and control room dose due to the containment air release pathway to remove the credit for the scrubbing of elemental iodine by containment spray. As stated above, this is expected to have minimal (on order of 0.05 Rem TEDE to MCR) impact on results. Waterford 3 will submit the revised AST LBLOCA results to the NRC by October 22, 2004.	X		10/22/04