FPL Energy
Seabrook Station

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MAY 5 2004

Docket No. 50-443
NYN-04039

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Reference: FPLE Seabrook Letter NYN-03061, Seabrook Station License Amendment Request 03-02, "Implementation of Alternate Source Term," dated October 6, 2003.

Seabrook Station
"Response to Request for Additional Information
Regarding License Amendment Request 03-02"

Enclosed is the FPL Energy Seabrook, LLC (FPLE Seabrook) partial response to requests for additional information associated with License Amendment Request (LAR) 03-02 received on January 28, 2004, March 1, 2004 and March 23, 2004.

A response to RAI 2 from the request for additional information dated March 1, 2004 and RAI 5, 6d1, 6d2, 6d4, 6e4, 6g and 6i from the request for additional information dated March 23, 2004 will be provided at a later date.

Should you have any questions concerning this response, please contact Mr. James M. Peschel, Regulatory Programs Manager, at (603) 773-7194.

Very truly yours,

FPL Energy Seabrook, LLC

Mark E. Warner
Site Vice President

A001

Cc: H. J. Miller, NRC Region I Administrator
V. Nerses, NRC Project Manager, Project Directorate I-2
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Mr. Bruce Cheney, Director
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OATH AND AFFIRMATION

I, Mark E. Warner, Site Vice President of FPL Energy Seabrook, LLC, hereby affirm that the information and statements contained within this response to the Request for Additional Information to License Amendment Request 03-02 are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed
before me this
5th day of May, 2004

Victoria Susan Robertson
Notary Public

VICTORIA SUSAN ROBERTSON
MY COMMISSION EXPIRES JAN. 15, 2008
NOTARY PUBLIC
NEW HAMPSHIRE

Mark E. Warner
Site Vice President

Enclosure to NYN-04039

Seabrook AST RAI Responses-First Set dated 1/28/04

RAI 1

The fission products which result from significant core damage can be limited from escaping the containment environment through chemical means. In particular, elemental iodine formed during the accident can be held in the liquid phase in the sump if the sump is at a pH of 7.0 or greater. Sump pH is dependent on the post-LOCA production of acids (Hydrolodic Acid, Hydrochloric Acid, and Nitric Acid) and the post-LOCA production and/or addition of bases (Cesium Hydroxide and Sodium Pentaborate).

1. The staff requests the licensee to provide additional details related to the assumptions used in calculating sump pH. Specifically, the staff requests the licensee to list the sources of post-LOCA acid generation affecting the sump pH and the mechanism of acid formation for each source (e.g., formation of HCL from the decomposition of cable insulation).

   In addition, the staff requests a time dependent list of the calculated post-LOCA acid generation and sump pH values similar to the following example table.

Time (hours)	HCL moles	HI moles	HNO_3	pH
2	102	.3	4	7.12

FPLE Response to RAI 1:

Following a LOCA, borated water is added to the containment sump from the Refueling Water Storage Tank (RWST) and the Reactor Coolant System. In order to maintain the required pH of the Containment Spray and the containment sump, sodium hydroxide from the Spray Additive Tank (SAT) is mixed with the borated water in the mixing chamber of the Refueling Water Storage Tank. The borated water from the Refueling Water Storage Tank and the sodium hydroxide from the Spray Additive Tank are added to the containment sump within one hour after a LOCA. The descriptions of the Containment Spray and Safety Injection systems are provided in the Seabrook Station UFSAR Section 6.2.2. Achievement of a sump pH greater than 7 occurs early in the time interval where processes leading to the formation of elemental iodine by radiolysis occur. Following this initial chemical addition/homogenation, the sump pH would only be impacted from the nitric acid produced from irradiation of water, the hydrochloric acid produced by irradiation of electrical cable insulation, and sump water temperature changes. Note that the analysis conservatively does not consider the addition of base forming components such as cesium hydroxide, which would increase pH.

The minimum containment sump pH was developed based on using the combination of boric acid and sodium hydroxide concentrations and tank volumes that would produce a minimum pH value. In addition to the boric acid, nitric and hydrochloric acids produced from irradiation of water, and electrical cable insulation were included. The nitric and hydrochloric acids were developed using the guidelines provided in NUREG/CR-5950, "Iodine Evolution and pH Control," using 30-day integrated doses in the sump and containment atmosphere at the stretch power uprate conditions.

The containment sump pH at 30-days was determined to be greater than 8.0 at a nominal temperature of 86°F. This is the lowest pH value expected to be found inside containment during the 30-day evaluation period after sump mixing and well above the minimum required value of 7 to control the re-evolution of iodine. The use of 86°F in the analysis is conservative because, in the highly buffered system of boric acid and caustic, the pH increases with increasing temperature.

Since all of the boric acid and sodium hydroxide is added to the containment sump within one hour, the moles of nitric (479.2 g-mol) and hydrochloric (1.6E3 g-mol) acids formed are based on a 30-day integrated dose and the 30-day pH value is well above 7, a time dependent sump pH with component inventories has not been provided. The pH would actually be higher, but less than the pH limit of 10.5, during the early time period because smaller quantities of nitric and hydrochloric acids would be produced.

The 30-day containment sump pH was determined with and without the additional acids produced from the irradiation of water and electrical cable insulation. The additional acid reduces the final pH by less than 0.1 pH units. Hydriodic acid was not considered in this evaluation since only small amounts are released from the core and are much less than the hydrochloric and nitric acid present.

RAI 2:

The staff requests the licensee to demonstrate how the sump pH will be maintained above 7 for the period of 30 days.

FPLE Response to RAI 2:

As discussed in FPLE's response to RAI 1 above, the ultimate sump pH at 30 days is well above the required value of 7 to control the re-evolution of iodine. The ultimate sump 30-day pH value is the minimum value for pH after depletion of the RWST and SAT in approximately 1 hour.

RAI 3:

If the sump pH was determined through the use of a computer code, the staff requests the licensee to identify the code and to detail the input and output of the code.

If the sump pH was determined through another analysis, the staff requests the licensee to provide details of the calculations including the assumptions.

FPLE Response to RAI 3:

A computer program was not used to develop the containment sump pH. As stated above, the pH value was determined at 30-days which is a lower value than expected in the containment sump at approximately 1 hour when recirculation begins. A manual calculation was performed to determine the pH value at 30-days using the inputs described in Table 1 below.

Table 1

Input Description	Value
Refueling Water Storage Tank Volume Released (Maximum Value @ 98°F)	428,000 gallons
Refueling Water Storage Tank Boron Concentration (Maximum Value)	2900 ppm
Spray Additive Tank Volume Released (Minimum Value @ 50°F)	8,520gallons
Spray Additive Tank Sodium Hydroxide Concentration (Minimum Value)	19 wt.%
Reactor Coolant System Mass	492,200 lbm
Reactor Coolant System Boron Concentration (Maximum Value)	4,000 ppm
Accumulator Volume (Maximum Value @ 100°F) – 4 Accumulators	6,596 gallons/accumulator
Accumulator Boron Concentration (Maximum Value)	2900 ppm
Mass of Electrical Cable Insulation[1]	50,000 lbm
Post-LOCA 30-Day Integrated Radiation Dose in Containment Sump water at SPU Conditions (Beta & Gamma)	3.3E07 rads
Post-LOCA 30-Day Integrated Radiation Dose in Containment Air at SPU Conditions (Beta & Gamma)	7.1E07 rads

Notes: 1. Moles of acid formed due to the decomposition of cable insulation are presented in the response to the first set of RAIs -RAI 1.

Seabrook AST RAI Responses-Second Set dated 3/1/04

RAI 1:

Provide the 1998 through 2002 hourly meteorological data used in ARCON96 calculations and joint wind speeds, wind direction and atmospheric stability distributions (jfd) used in the PAVAN calculations. If these data have already been provided on the docket, please cite an appropriate reference. Please specify if the jfd data are formatted as discussed on Page 14 of Enclosure 2 to NYN-03061 (Enclosure 2) provided by letter dated October 6, 2003, or if they are the reformatted data input to the PAVAN calculations.

FPLE Response to RAI 1:

Enclosed is the 1998 through 2002 meteorological data used in the ARCON96 calculations. Also enclosed are the joint wind speeds, wind directions and atmospheric stability distributions (jfd) used in the PAVAN calculations. The jfd data used to determine the offsite X/Q values were reformatted for the PAVAN code input in the PAVAN input file (seabrookinput.dat). The process to reformat the data from the raw jfd files is discussed on page 14 of Enclosure 2 to NYN-03061.

The lower meteorological measurement height is 10.05 m and the upper is 60.66 m. The windspeed values are in miles per hour.

RAI 2

Page 15 of Enclosure 2 states that the exit velocity from the MSSVs is greater that the 95 percentile wind speed for the first 2 ½ hours of the events during which a release is postulated to occur. What are the estimated exit velocities, flow rates, and pressures as a function of time for the MSSVs and ASDVs during this interval? What is the basis for the estimates? The 95 percentile wind speeds are estimated to be 16.72 and 16.81 miles per hour. How were these values estimated?

FPLE Response to RAI 2:

The response is currently being developed and will be submitted at a later date.

RAI 3:

With regard to the ARCON96 relative concentration (X/Q) estimates provided in Table 1.8.1-2 of Enclosure 2, were all of the calculations based upon the same assumptions, other than 1) the differences noted in Table 1.8.1-1, 2) reduction by a factor of two when either the east or west intake was the postulated receptor location, or 3) when the diesel building intakes were assumed receptors and the X/Q value is a weighted average of the two intakes. In the third case, are the input values given in Table 1.8.1-1 for the more limiting intake and inputs for the less limiting intake not provided? Are the inflow rates assumed to be equal? Were any releases assumed to be diffuse or to have a vent flow? If so, what were the inputs? Are values provided in Table 1.8.1-2 applicable to loss of offsite power and single failure scenarios?

FPLE Response to RAI 3:

All of the calculations performed on the X/Q estimates are based on the same assumptions other than the items listed in this RAI question.

The only additional differences not listed in Table 1.8.1-1 are: (1) the cases with releases from the plant vent and containment building which credit building wake, and (2) the cases with the releases from the RWST with the diesel building intakes as the receptors which credit building wake in the ARCON96 case runs. The building area used for the plant vent and containment building release cases in the ARCON96 files is 1,506 m^2. The building area used for the RWST to diesel building intake cases is 337 m^2.

For the third case as stated in the RAI, where a weighted average of the two diesel building intakes is used, the values given in Table 1.8.1-1 are for the more limiting intake, and the less limiting intake values are not provided. The inputs for the case from the RWST to the non-limiting diesel intake are the same as the case to the limiting intake except that the distance is 52.4 m and the direction is 137°. The inflow rates for the diesel building intakes are assumed to be equal as previous analyses have shown.

Conservatively, no releases were assumed to be diffuse and no releases were assumed to have a vent flow.

The values provided in Table 1.8.1-2 are applicable to loss of offsite power and single failure scenarios.

Seabrook AST RAI Responses-Third Set, dated 3/23/04

RAI 1:

Regarding the proposed technical specification change in the definition of "dose equivalent I-131," Seabrook uses the thyroid dose as the basis of the proposed change. This definition finds use in the Reactor Coolant System (RCS) and secondary specific activity technical specifications. The purpose of those technical specifications is to control the actual specific activities to levels less than those which would exceed the initial assumptions made in the radiological consequences analyses. Previously, those analyses determined whole body and thyroid does, consistent with the dose guidelines in 10CFR100.11. However, with the proposed implementation of the Alternative Source Term (AST), the total effective dose equivalent (TEDE) criteria supercede the whole body and thyroid dose. The staff has not required licensees to revise this definition. Since you have proposed a change, please provide a justification for the use if thyroid dose conversion factors when the effective factors provided in Federal Guidance Report (FGR) 11 Table 2.1 would be more appropriate.

FPLE Response to RAI 1:

In the Seabrook dose calculations, the dose conversion factors referenced in the Technical Specification definition of dose equivalent I-131 (D.E. I-131) are used to adjust the initial primary coolant iodine activities.

The primary coolant iodine activities adjusted to 1.0 ηCI/gm D.E. I-131 based on the thyroid dose conversion factors are:

Primary Coolant Iodine Activities Based on FGR-11 Thyroid Dose Conversion Factors

Isotope	Adjusted µCi/gm
I-131	0.7727
I-132	0.2813
I-133	1.2363
I-134	0.1793
I-135	0.6800

Use of FGR-11 Table 2.1 effective dose conversion factors, results in the following iodine activities (after adjusting the initial primary coolant iodine activities to 1.0 ηCI/gm D.E. I-131):

Primary Coolant Iodine Activities Based on FGR-11 Effective Dose Conversion Factors

Isotope	Adjusted μCi/gm
I-131	0.7562
I-132	0.2753
I-133	1.2099
I-134	0.1754
I-135	0.6655

As can be seen by examining the two tables, using the thyroid dose conversion factors in the definition of D.E. I-131 results in higher iodine concentrations in the primary coolant. Thus, using the thyroid dose conversion factors produces a more conservative determination of the primary coolant iodine activity for use in the dose calculations.

In order for Technical Specifications to be consistent with the conservative analytical basis, the definition of dose equivalent I-131 (D.E. I-131) was established based on the thyroid dose conversion factors. This approach is consistent with other previously approved Alternative Source Term submittals. For example, Page 22 of the Safety Evaluation for Shearon Harris Nuclear Power Plant Unit 1 Amendment No. 107 to Facility Operating License No. NPF-63 dated October 12, 2001 (ADAMS Accession No. ML012830516) states:

"Revise TS 1.11 definition of dose equivalent iodine-131 to read in part,

The thyroid dose conversion factors used for this calculation shall be those listed in the International Commission on Radiological Protection (ICRP), "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Volume 3, No. 1-4, 1979 (or equivalently, Federal Guidance Report No. 11, "Limiting Values of Radionuclides Intake and Air Concentration and dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA 520/1-88-020, September 1988)."

<u>RAI 2</u>

For the gaseous and waste system failure events, Seabrook proposes to use the current licensing basis criterion of a "small fraction of the guidelines." The staff did not address these two events in Regulation Guide 1.183 since these events are not likely to result in core damage. Therefore, no AST specific dose criteria were provided. Nonetheless, the staff notes that the Standard Review Plan Sections 15.7.1 and 15.7.2 and 15.7.4 impose acceptance criteria from Branch Technical Position 11-5. These in turn derived from 10CFR Part 20 rather than Part 100. The staff's original Safety Evaluation Report (SER) does not appear to address the radiological consequences of these events. Please briefly describe the basis of the Seabrook Offsite Dose Calculation Manual (ODCM) controls that limit the contents of these tanks. Please explain any significant differences between these basis and the acceptance criteria you are proposing in the License Amendment Request (LAR).

FPLE Response to RAI 2:

Reg. Guide 1.183 does not provide guidance for the dose limits for a failure in the Radioactive Gaseous Waste System or the Radioactive Liquid Waste System; therefore, the current Seabrook licensing basis was investigated as the most appropriate basis to establish the acceptance criteria for these events. The applicable portions of the current UFSAR sections discussing acceptance criteria for these events (Sections 15.7.1.4 and 15.7.2.4) were part of the FSAR (Amendment 53) that was originally submitted to support the Seabrook Station license and associated Seabrook Safety Evaluation Report (NUREG-0896, "Safety Evaluation Report related to the operation of Seabrook Station, Units 1 and 2," dated March 1986). The July 1981 version of the Standard Review Plan (SRP) referenced in the Seabrook Safety Evaluation Report (NUREG-0896, "Safety Evaluation Report related to the operation of Seabrook Station, Units 1 and 2," dated March 1986) had deleted the 15.7.1 and 15.7.2 sections indicating the SRP did not classify these events as Design Basis Accidents (DBAs). The Radioactive Gas Waste System (RGWS) failure acceptance criteria established by Section 15.7.1.4 of the current Seabrook UFSAR concludes only that the consequences are "below the values specified in 10 CFR Part 100." Based upon the statement in Section 15.7.2.4 of the Seabrook Station UFSAR, the doses from the Liquid Waste System Failure event are concluded to be "within a small fraction of the 10 CFR Part 100 Guidelines" which is interpreted to mean less than 10% of the limit. Therefore, the submitted RGWS failure analysis also assumes the more conventional "small fraction" criteria of 10% of the limit to be consistent with the criteria established for the liquid waste system releases.

Note that the latest proposed revision to BTP 11-5 endorses the "small fraction of the 10 CFR Part 100 guidelines" as the criteria for this event. This criterion is also conservative with respect to the dose limit applied for the Kewaunee Nuclear Plant where selection of the WGDT Rupture dose consequences acceptance criteria for the EAB and LPZ was based on Fuel Handling Accident acceptance criteria (6.3 rem). This precedent is established in Section 8.2 of the Kewaunee Nuclear Power Plant AST Engineering Report submitted March 19, 2002, in Section 8.2 of the revised Kewaunee AST Engineering Report submitted on October 21, 2002 (ADAMS Accession No. ML023040302) and subsequent Issuance of Amendment (IA) and Safety Evaluation (SE) issued March 17, 2003 (ADAMS Accession No. ML030210062).

The Radioactive Liquid Waste System is designed to maintain, during normal operation, the radioactivity content of liquid effluents from the Seabrook site within the concentration limits expressed in 10CFR20. The original plant analysis was based on an activity inventory of 1% failed fuel. The Alternative Source Term analysis also assumes the maximum possible source term is in the tank and is available for release. The contents of the Radioactive Liquid Waste System are controlled by plant operations. There are no specific restrictions for discharge to the Radioactive Liquid Waste System. The Seabrook Station ODCM ensures that the dose, at any time, at and beyond the site boundary from effluent will be within the annual dose limits of 10 CFR 20 to unrestricted areas. These releases do not include accident conditions; therefore, these limits apply to normal plant operation and its effluents and not for accident conditions.

The Radioactive Gaseous Waste System provides sufficient holdup and control of gaseous releases. Design is based on continuous operation with reactor coolant radioactivities associated with 1 percent failed fuel at rated thermal power. The original plant analysis was based on a source term of noble gas inventory from five carbon delay beds and of a activity inventory of 1% failed fuel. The original analysis also assumes 100% of the noble gas would be released to the environment over two hours. The Alternative Source Term analysis also assumes the maximum possible source term is in the tank and is available for release. The contents of the Radioactive Gaseous Waste System are controlled by plant operations. There are no specific restrictions for discharge to the Radioactive Gaseous Waste System. The Seabrook Station ODCM ensures that the dose, at any time, at and beyond the site boundary from effluent will be within the annual dose limits of 10 CFR 20 to unrestricted areas. These releases do not include accident conditions; therefore, these limits apply to normal plant operation and its effluents and not for accident conditions.

RAI 3:

With regard to control room emergency ventilation actuation, Seabrook has assumed a 30 second delay in actuation for all analyzed accidents. In paragraph 1.6.3, Seabrook states that this actuation is based on high radiation being detected in the remote air supply piping. On page 20 of 94, it is stated that for the Loss of Coolant Accident (LOCA), a containment high pressure signal actuates isolation and that the 30 seconds are provided for diesel generator start time and damper actuation and positioning time. Please explain how the assumed 30 second delay is conservative for all accidents, considering the response considerations identified by FPL, but also how the time for the input activity to ramp up to the alarm set point level and the impact of differences in accident specific radionuclide effluent mixes on monitor response are considered.

FPLE Response to RAI 3:

The current licensing basis for control room habitability is described in Seabrook Station UFSAR Section 6.4. and the emergency mode of operation is described in UFSAR section 6.4.3.2.

The control room ventilation system will be isolated if high radiation is sensed at either intake, a safety injection signal (S signal) occurs, or the system is manually isolated. Once a control room ventilation isolation signal is generated, the following occurs:

- The normal operating intake fans stop and their discharge dampers close
- The emergency intake and cleanup filter fans start and their dampers open.

The response time for the control room to switch over to recirculation/filtration is 5 seconds (after receipt of a high radiation signal, or S signal). The radiation monitors are located directly in the entrances of the control room remote air intakes. The distance from the location of the radiation monitors to the ventilation isolation dampers is quite long. Based on the air intake flow rate and based on the dimensions of the intake, the minimum transit time from a radiation monitor to an isolation damper is 8.3 seconds. This transit time is longer than the response time for the isolation of the control room. Therefore, the control room will be isolated in 5 seconds, and the isolation system will prevent any non-filtered air from entering the Control Room.

The response times for the Engineered Safety Features Actuation System (ESFAS) are provide in Technical Requirement Manual, TR 2. The response times are summarized as follows:

Event	CBS Emergency Fan/Filter Actuation Time
Control Room Intake Hi Radiation	< 5 seconds
Containment Pressure - Hi-1 (S-signal)	< 5 seconds
Pressurizer Pressure - Low (S-signal)	< 5 seconds

For the Alternative Source Term analysis, a bounding and conservative isolation time of 30 seconds was used in order to provide design margin.

As per Section 12.3 and Table 12.3-16 of the Seabrook Station UFSAR, the air intake radiation monitors are GM-type detectors that are located directly in the duct air stream. These detectors do not depend on air pumps to provide an air sample for analysis, nor are they shielded. The setpoint for these monitors is 2 times background (2 x 0.5 = 1 mR/hr) or 100 cpm; thus, for any dose significant event, it can be assumed that these monitors will initiate a control room isolation signal. These detectors have a low sensitivity and will respond to low levels of radionuclide effluent mixes. Thus, there is no delay or ramp up time for the exposure of the radiation monitors to the release.

The assumption that a high radiation signal will be generated due to the low setpoint of the radiation monitors can be confirmed by examining the average whole body dose rates due to noble gas at the entrance to the most limiting control room air intake for the first 30 seconds of the Seabrook Station events. The RADTRAD-NAI dose calculation model assumes instantaneous transport of releases from the release point to the receptor point; thus, there is no delay or ramp up time for the exposure of the radiation monitors to the release:

Event	Whole Body Dose Rate at Entrance to Most Limiting Air Intake Due to Noble Gas (average over first 30 seconds) mrem/hr
MSLB	1.27
SGTR	264.83
Locked Rotor	260.93
Letdown Line Break	95.15
Radioactive Gaseous Waste System Failure	18471.16

RAI 4a:

Regarding the control room unfiltered inleakage assumptions:

4.a For those events in which the 20 cfm door leakage is not assigned to a particular infiltration point, is the value included in the inleakage values shown in Table 1.6.3-1?

FPLE Response to RAI 4a:

Yes, the values listed under "Unfiltered Inleakage (Total)" in Table 1.6.3-1 are the total assumed unfiltered inleakage including the 20 cfm door leakage.

RAI 4b:

Regarding the control room unfiltered inleakage assumptions:

3.4.b In its Generic Letter 2003-01 response, Seabrook reported preliminary results for the Seabrook inleakage testing. Please confirm that the final test results are bounded by the minimum inleakage assumption shown in Table 1.6.3-1.

FPLE Response to RAI 4b:

The final tracer gas test results as follows:

Mode Tested	Unfiltered In-leakage
Train A	8 ± 11 SCFM
Train B	14 ± 22 SCFM

Based on the above, the final test results are bounded by the minimum in-leakage assumptions shown in Table 1.6.3-1 of Enclosure 2 of NYN-03061.

RAI 5

Regarding Table 1.8.2-1, the staff is of the opinion that only the 0-2 hour exclusion area boundary (EAB) X/Q value has applicability to the radiological consequence calculations that determine the worst two hour EAB dose. If the values for time periods beyond two hours were used in the analysis of the worst two hour dose, please explain how the values were used and why this approach should be considered acceptable.

FPLE Response to RAI 5:

The response is currently being developed and will be submitted at a later date.

RAI 6a

Regarding the LOCA analysis:

3.6.a in paragraph 2.1.2.4, Seabrook states that they are assuming an aerosol deposition rate of 0.1 per hour, based on Industry Degraded Core Rulemaking Program (IDCOR) Technical Report 11.3. RG 1.183 Regulatory Position 3.4 identifies NUREG/CR-6189 is an acceptable approach. Since this parameter is somewhat dependant on plant parameters, the staff's prior approval of 0.1 per hour for another licensee may not be relevant to Seabrook. Please provide a Seabrook specific justification for your proposed deviation of this guidance.

FPLE Response to RAI 6a:

The IDCOR value and methodology used are applicable to Seabrook Station are reasonable and conservative when compared to NUREG/CR-6189. Table 34 of NUREG/CR-6189 presents decontamination coefficients for design basis accident aerosol deposition. These decontamination coefficients are presented as a function of thermal power, time range and release phase. Table 36 of NUREG/CR-6189 presents correlations to model these decontamination coefficients as a function of thermal power, time range and release phaseNUREG/CR-6604 (RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation) Table 2.2.2.1-1 presents correlations for determining the same natural deposition aerosol decontamination coefficients as a function of power, time and release phase (same as Table 36 of NUREG/CR-6189, but sums the gap and early in-vessel release phases). Thermal power is the only parameter that is varied in this table. The following is an excerpt from page 6 of the Safety Evaluation for Indian Point Nuclear Generating Unit No. 2 Amendment No. 211 to DPR-26 dated July 27, 2000 (ADAMS Accession No. ML003727500):

"Lower bound (10 percentile) natural processes decontamination coefficients for radiological design-basis accidents were identified in Table 34 of NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," July 1996. The natural processes aerosol removal model in the staff's confirmatory analysis code RADTRAD is based on NUREG/CR-6189. Based on Table 34, the staff finds the sedimentation removal coefficient of 0.1 hr^{-1} to be reasonable."

This conclusion applies to Seabrook Station since the analyzed Seabrook thermal power level (3659 MWt) is greater than the analyzed Indian Point Unit 2 thermal power level (3216.5 MWt) and the values of decontamination coefficients in Table 2.2.2.1-1 of NUREG-6604 increase with thermal power for these two values.

Several other Safety Evaluation Reports also support the use of 0.1 hr^{-1}. The following is an excerpt from page 5 of the Safety Evaluation for Kewaunee Nuclear Power Plant Amendment No. 166 to DPR-43 dated March 17, 2003 (ADAMS Accession No. ML030210062):

"The fission products in the containment atmosphere following the postulated LOCA [are] mitigated by natural deposition processes and by the containment spray system (CSS). The licensee assumed a radioactive aerosol removal rate of 0.1 per hour in the containment atmosphere. This removal credit is taken after the CSS operation is terminated. The NRC staff finds 0.1 per hour aerosol removal rate to be reasonable (within the 85 percent of the uncertainty distribution) based on [the] study published in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor containments," and [it] is therefore acceptable." The Seabrook Station analyzed thermal power (3659 MWt) is also greater than the Kewaunee Nuclear Power Plant analyzed thermal power (1851.3 MWt). As mentioned previously, the values of the decontamination coefficients in Table 2.2.2.1-1 of NUREG-6604 increase with thermal power for these two values.

RAI 6b:

Regarding paragraphs 2.1.2.11, .12, .15, please confirm the staff's understanding that paragraph 2.1.2.11 and 2.1.2.12 apply to 40 percent L_a leakage and that the drawdown does not change the 60 percent bypass assumption.

FPLE Response to RAI 6b:

The staff's understanding that paragraph 2.1.2.11 and 2.1.2.12 of Enclosure 2 to NYN-03061 apply to 40% La leakage and that the drawdown does not change the 60 percent bypass assumption is correct.

RAI 6c

Regarding paragraph 2.1.2.15, what is the basis of the 40-60 split in containment leakage.

FPLE Response to RAI 6c:

The 60/40 split is based on the Seabrook Station current Licensing Basis and is specified in UFSAR Section 15.B.2.IID as 0.60 La for the Containment Enclosure Emergency Exhaust Filter Bypass Fraction (conservative analysis). The maximum allowable leakage (L_a) from the containment structure following an accident is 0.15 percent of the mass of its atmosphere per day. This would occur at a maximum pressure. The direct leakage to the environs of radioactive contaminants from the containment is within the guidelines of 10 CFR 100.

Although a containment enclosure emergency cleanup system has been provided to minimize leakage to the environs, a significant number of lines penetrate the containment and terminate in areas not treated by this cleanup system. Therefore, leakage attributed to penetrations and isolation valves, requiring Type B and Type C Test per 10 CFR 50, Appendix J, is conservatively assumed to bypass the cleanup system, and is limited to 60% of the maximum allowable leakage (L_a). The remaining 40% of L_a is assumed to enter the containment enclosure, and to be treated by the containment enclosure emergency cleanup filtration system, prior to release. Seabrook Station tests to the design basis values in accordance with Appendix J.

RAI 6d1 and 6d2:

Regarding paragraph 2.12.19 through 2.1.2.22, the staff cannot find FPL's treatment of emergency core cooling system (ECCS) leakage acceptable without additional supporting justification for the following deviations from guidance:

- Regulatory Position 5.3 states that "with the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in liquid phase." Seabrook has stated the "with the exception of the non-particulate iodines, all radioactive materials in the recirculating liquid are assumed to be retained in liquid phase."

- Regulatory Position 5.4 and 5.5 provide that the flashing fraction is to be based on the fraction of the total iodine in the liquid. Seabrook proposes that 100% of the non-particulate iodine becomes airborne, but none of the particulate iodine becomes airborne.

- Regulatory Position 5.6 states that the radioiodine available for release is assumed to be 97 percent elemental and 3 percent organic. Seabrook states that the temperature and pH history of the sump and refueling water storage tank (RWST) are considered in determining the chemical form of iodine.

The staff structured these regulatory positions to be deterministic and conservative in order to compensate for lack of research into iodine speciation beyond the containment and the uncertainties of applying laboratory data to the post accident environment of the plant. Regulatory Position 5.5 does state that a smaller flash fraction could be justified based on the actual sump pH history and area ventilation rates. The staff believes that Seabrook has not provided sufficient data for the staff to find its proposed treatment of ECCS leakage adequately conservative. Please provide a quantitative justification for your assumptions including, but not limited to, the following information:

6.d.1) A full description of the iodine speciation analysis that supports your assumptions, including methodology, assumptions , input data.

6.d.2) A discussion of how the iodine speciation may change as the containment sump water is circulated through the ECCS components and piping and out to the RWST.

FPLE Response to RAI 6d1 and 6d2:

The response is currently being developed and will be submitted at a later date.

RAI 6d3

A discussion of the impact of all possible post accident liquid inputs to the RWST with other sources of water.

FPLE Response to RAI 6d3:

For the Design Basis LOCA with ECCS leakage and emergency recirculation functioning, no additional water would be added to the RWST based on the Seabrook Station current Licensing Basis and operating procedures and are not required by Licensing Basis.

RAI 6d4:

A discussion on how the iodine speciation might change as the ECCS leakage is sprayed out of a leak, or streams across a floor into a building sump.

FPLE Response to RAI 6d4:

The response is currently being developed and will be submitted at a later date.

RAI 6e1, 6e2, and 6e3:

On page 20 of 94, the basis for the air flow rate is provided. Please address the following:

RAI 6e1

The air flow is based on the average daily temperature swing of 18.2 degrees. This temperature swing appears low for a summer day. Please explain how this value was determined and why it should be considered adequately conservative.

RAI 6e2

Was evaporation of the RWST water considered as a contributor to the air flow rate?

RAI 6e3

Since the iodine partition is the ratio of the vapor pressure of the iodine in the liquid and gas phases in the RWST, please discuss the impact of tank pressure changes associated with the diurnal temperature swings.

FPLE Response to RAI 6e1, 6e2, and 6e3 Response

As presented on page 14 of Enclosure 2 to NYN-03061, the average daily temperature swing used in the RWST releases is determined from the 2001 ASHRAE fundamentals handbook. The value of 18.2 °F for the dry bulb temperature swing from Portsmouth NH was used as the closest listed location at the Seabrook site. The modeling of the potential releases from an RWST over an extended period of time is subject to a number of subtle factors that can not be reasonably computed or modeled. The approach used in this analysis was to impose some very conservative assumptions to bound the major factors and use engineering judgment to conclude that these major conservatisms provided adequate margin to cover the more subtle phenomenon taking place in the tank as discussed below.

The assumed daily expansion and contraction of the vapor in the RWST contributes substantially to the volumetric, and therefore, the radiological release from the RWST. Without the daily expansion contraction cycle the release would be limited to the small volume of vapor/air mixture displaced by backleakage flow. Therefore the conservative treatment of the daily air/vapor expansion and contraction is in fact a major conservatism in the overall computation of the dose contribution from this source. First, the assumption that the internal temperature of the tank will be subject to the same daily temperature swings as experienced for the outside air is itself very conservative. This assumption ignores the substantial thermal mass of the tank walls and the liquid in the tank. Furthermore, the tank itself is housed inside a sizable building which will substantially mitigate the daily temperature swing to which the tank is subjected. Therefore, even the assumed 18.2 °F temperature swing on the outside of the tank is very conservative. Rather than attempting to model or estimate the impact of these factors, the analysis simply made the very conservative assumption that the tank internal temperature could swing as high as the daily outside air temperature swing. Again, considering the extent to which this assumption is conservative and the sensitivity of the dose assessment on this parameter, this constitutes a major conservatism in this dose contribution.

Evaporation is not specifically computed as an individual contribution to the air flow rate from the tank. However, the effect of evaporation is essentially included in the air flow rate because the air/vapor space in the tank was assumed to achieve an equilibrium vapor pressure with the liquid phase. Therefore the previously discussed air/vapor release from the tank due to daily heating and cooling includes water vapor evaporated to maintain that equilibrium. This is consistent with the modeling of the release from the RWST which assumes that the activity in the vapor space is in equilibrium with the activity in the liquid space based upon the Iodine partition factor calculated for liquid temperature. In this respect, this analysis does account for the effect of evaporation on the release from the RWST. Furthermore, the analysis does not assume any heat losses from the tank either through evaporation or conduction throughout the 30 days of the event. This assumption is imposed in order to establish a conservatively high RWST water temperature that provides a conservative partition factor for elemental iodine between the liquid and vapor space. The partition factor is very sensitive to temperature and a more realistic model considering the heat losses from the tank over several days would produce substantially lower elemental iodine concentrations in the tank vapor space and correspondingly lower activity releases. This no heat loss assumption represents another major conservatism inherent in the analysis.

To maximize the release from the tank, no flow restriction is assumed to be credited through the tank vent path and the tank is assumed to remain at atmospheric pressure. If even slight pressurization of the tank could be credited, the daily heating cooling cycle release flow could be reduced or eliminated and the dose would be significantly reduced. This would be expected to more than offset the impact that such a slight pressure change would have on the iodine partition factor in the tank.

As demonstrated by the discussion above, the complexity and uncertainties in the analysis of the dose contribution from the RWST have been addressed by imposing some very conservative simplifying assumptions on aspects of the analysis that are difficult to quantitatively assess. As discussed in other responses to questions concerning the iodine species in the sump and RWST water, this very conservative treatment of uncertain conditions is also inherent in the assumptions and modeling used in those aspects of the analysis. The imposition of these multiple combined conservatisms provide confidence that the overall assessment of the dose from this aspect of the event is conservative.

<u>RAI 6e4</u>

As noted above in question 6d, the staff questions the iodine fraction value.

<u>FPLE Response to RAI 6e4:</u>

The response is currently being developed and will be submitted at a later date.

<u>RAI 6f</u>

On page 21 of 94, a mixing rate of two turnovers per hour is assumed. Regulatory position 3.3 provides this as a default assumption, if adequate flow exists between these two regions. Please briefly describe the basis for assuming that this flow will exist between the sprayed and unsprayed regions.

<u>FPLE Response to RAI 6f</u>

Per Seabrook Station UFSAR Section 15.6.5.4.a, "Radiological Consequences – Spray Removal Analysis", The mixing rate is 2 hr^{-1} from the unsprayed region. The UFSAR states the following:

"The effectiveness of the Containment Spray System has been evaluated using a two region spray model which assumes that 85.4 percent of the containment volume is directly wetted by the spray solution. Mixing between the sprayed and unsprayed region is assumed to occur at a rate of 13,000 cfm, corresponding to a mixing rate of is 2 hr^{-1} from the unsprayed region."

The two air changes an hour are attributable strictly to natural convection.

<u>RAI 6g</u>

On page 21 of 94, the maximum decontamination factor (DF) for elemental and particulate iodines are discussed. Please explain how the initial maximum airborne iodine concentration in the containment was determined for this determining DF.

<u>FPLE Response to RAI 6g</u>

The response is currently being developed and will be submitted at a later date.

RAI 6h:

Table 2.1-1 identifies the containment enclosure drawdown time for the LOCA as 4.5 minutes (270 seconds). Table 2.6- identifies the draw down time for the rod control cluster assembly (RCCA) ejection accident as 360 seconds. Appendix A of the Seabrook Updated Final Safety Analysis Report (UFSAR) states that filtration credit is not assumed for the first eight minutes. Please explain the difference in these values. What is the value of the acceptance criteria for surveillance testing of this system safety function?

FPLE Response to RAI 6h:

As described in Seabrook Station UFSAR section 6.2.3.5, the Containment Enclosure Emergency Air Cleanup System is automatically initiated on a 'T' (containment isolation Phase A) signal. The required response time for the Containment Enclosure Emergency Air Cleanup System is provided in Technical Specification 3/4.6.5. The required response time is summarized as follows:

Event	CBS Emergency Fan/Filter Actuation (Drawdown) Time
Containment Isolation Phase 'A' (T-signal)	≤ 4 minutes

For the Alternative Source Term analysis, a bounding and conservative drawdown time of 6 minutes (360 seconds) was used in order to provide design margin for the RCCA ejection accident.

The value of the acceptance criteria for surveillance testing of this system is to evacuate to –0.25 in H_2O in less than four minutes. This acceptance criteria is only for drawdown and does not include the signal time.

The value of 8 minutes in Appendix B of the UFSAR is in error and will be corrected through the Seabrook Station Corrective Action Program.

RAI 6i

Section 2.1.2.13 addressed Regulatory Position 4.3 and states that the containment enclosure emergency air cleaning system is capable of maintaining a negative pressure with respect to high wind speeds. UFSAR sections 6.5.1.1 and 6.5.1.3 are cited. UFSAR Section 6.5.1.3 states, "The calculated wind speed that would initiate building exfiltration is 17 miles per hour. At this or at a higher wind speed, any exfiltration will be adequately dispersed." Please explain the basis of this conclusion. What is meant by adequately dispersed? What is the 95 percentile wind speed at Seabrook? What impact does this windspeed have on the time to reach 0.25 inch water gage (WG).

FPLE Response to RAI 6i:

The response is currently being developed and will be submitted at a later date.

RAI 6j:

The Seabrook UFSAR provides an analysis of the consequences of post accident venting as a backup to the redundant hydrogen recombiners. This analysis was not address in the submittal. Is it Seabrook's intent to remove this analysis from the licensing basis? If not, why was this component of LOCA not address in the license amendment request?

FPLE Response RAI 6j:

The Combustible Gas Control System meets the redundancy and power source requirements for engineered safety features. No single failure will incapacitate the system. The Alternative Source Term Analysis credited the hydrogen recombiners for hydrogen control and considered post accident venting of hydrogen as a backup to the redundant hydrogen recombiners, and as such, beyond the design basis accident. Seabrook Station is currently pursuing a License Amendment Request to eliminate hydrogen control from the Technical Specifications and the post accident venting will be addressed in this effort. The proposed Technical Specification changes support implementation of the revisions to 10CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors," that became effective on October 16, 2003.

RAI 7

Regarding the main steam line break analysis, Table 2.3-1 lists an RCS mass of 539,037 lbm. Table 2.3-4 lists an RCS mass of 505,000 lbm. Table 2.6-1 lists a minimum RCS mass of 434,000 and a maximum mass of 539,037 lbm. While the staff understands minimum and maximum values may be used to maximize doses, it is not clear why the RCS mass assumed in establishing the iodine appearance rate was assumed to be 505,000 lbm. Please explain the basis for this assumption.

FPLE Response to RAI 7

This RAI response applies to RAIs 7, 8a, and 10 for Main Steam Line Break, Steam Generator Tube Rupture, and Letdown Line Break, respectively. The dose calculations for the AST analyses used the maximum RCS mass for calculations where the entire activity of the RCS was released or the release contribution from the initial RCS activity was being computed. The minimum RCS mass was used for cases where fuel failure activity was released into the RCS and mixing with the inventory in the pressurizer could not be assured.

For the concurrent Iodine spiking cases, an RCS mass of 505,000 lbm was originally used to establish the RCS initial equilibrium iodine concentrations documented in the Seabrook Station UFSAR which are subsequently used in this analysis to determine the appearance rate. To be consistent with that basis, the appearance rate calculation (source term) uses the same RCS mass as was assumed to establish the initial concentrations. To be consistent with the source term, the RADTRAD model uses the same RCS mass as was used to generate the source term.

The 1.0 μCi/gm dose equivalent I-131 equilibrium iodine activities were established by scaling the relative equilibrium iodine isotopic concentrations in the coolant presented in Table 11.1.1 of the Seabrook Station UFSAR to achieve the Technical Specification limit of 1.0 μCi/gm dose equivalent I-131. The value of 505,000 lbm obtained from UFSAR Table 11.1-3 was determined to be an appropriate RCS mass value for the concurrent spiking calculations.

RAI 8a:

With regard to steam generator tube rupture analysis:

Regarding the steam generator tube rupture analysis, Table 2.4-1 lists an RCS mass of 539,037 lbm. Table 2.4-4 lists an RCS mass of 505,000 lbm. Table 2.6-1 lists a minimum RCS mass of 434,000 and a maximum mass of 539,037 lbm. While the staff understands minimum and maximum values may be used to maximize doses, it is not clear why the RCS mass assumed in establishing the iodine appearance rate was assumed to be 505,000 lbm. Please explain the basis for this assumption.

FPLE Response to RAI 8a:

See FPLE Response to RAI 7 above.

RAI 8b

In paragraph 2.4.2.12, please clarify the phrase, "...without flashing for all steam generators..." as used in the first bullet. The use of "all" appears to be in conflict with the second bullet.

FPLE Response to RAI 8b:

In Section 2.4.2.12, the correct phrasing for the first bullet is "... without flashing for all ***intact*** steam generators...".

RAI 8c

The Table 1.8.1-3 entry for steam generator tube rupture (SGTR) uses language different from that for the main steam line break (MSLB), locked rotor, or RCCA ejection events. It appears that this difference in language provides for the factor of five plume rise reduction to be applied to noble gas releases for the entire eight hour release duration rather than 2.5 hours. If this is Seabrook's intent, please provide a justification for this assumption.

FPLE Response to RAI 8c:

The phrasing used to describe plume rise credit in Table 1.8.1-3 is confusing. When credit for plume rise is taken, it is only applied for the first 2.5 hours of the event. The descriptions used in Table 1.8.1-3 could be interpreted to imply that the Table 1.8.1-2 X/Qs include the factor of 5 reduction for plume rise; however, the X/Qs listed in Table 1.8.1-2 do not include a factor of 5 reduction for plume rise. For the steam generator tube rupture and main steam line break events, the factor of 5 reduction for plume rise was not applied to the noble gas release. In addition, for the main steam line break event, all of the noble gas was assumed to exit from the steam line break location.

RAI 9a

With regard to the RCCA ejection analysis:

Please respond to Questions 6a through c and 6h in the context of the RCCA ejection event.

FPLE Response to RAI 9a:

See FPLE Response to RAIs 6a, 6b, 6c and 6h.

RAI 9b

Please confirm the staff's understanding that the 0.375 percent full centerline melt is referenced to the entire core and not only that fraction of the core that exceeds departure from nucleate boiling (DNB)

FPLE Response to RAI 9b:

The NRC staff's understanding is correct. The 0.375 percent fuel centerline melt was applied to the entire core.

RAI 10

Regarding the letdown line break analysis, Table 2.7-4 lists an RCS mass of 505,000 lbm. Table 2.7-1 lists a minimum RCS mass of 434,000 lbm and a maximum mass of 539,037 lbm. While the staff understands minimum and maximum values may be used to maximize doses, it is not clear why the RCS mass assumed in establishing the iodine appearance rate was assumed to be 505,000 lbm. Please explain the basis for this assumption.

FPLE Response to RAI 10:

See FPLE Response to RAI 7 above.

RAI 11

Table 2.9-1 refers to non-existent Tables 2.10-2 and 2.10-3. Please confirm the staff's understanding that Table 2.9-2 is the intended reference.

FPLE Response to RAI 11:

The NRC staff's understanding is correct. Table 2.9-2 is the intended reference for the RLWS release inventory in Table 2.9-1. Additionally, Table 2.8-2 (versus Table 2.9-2) is the intended reference for the RGWS release inventory in Table 2.8-1.