February 24, 2005

Mr. Mark E. Warner, Site Vice President
c/o James M. Peschel
Seabrook Station
FPL Energy Seabrook, LLC
PO Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: ALTERNATIVE SOURCE TERM (TAC NO. MC1097)

Dear Mr. Warner:

The Commission has issued the enclosed Amendment No. 100 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1 (Seabrook), in response to your application dated October 6, 2003, as supplemented by letters dated May 5, May 24, July 8, September 13, 2004, and January 13, 2005.

The amendment revises the Seabrook licensing basis to incorporate a full-scope application of an alternative source term methodology in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Victor Nerses, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures: 1. Amendment No. 100 to License No. NPF-86
2. Safety Evaluation

cc w/encls: See next page

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SUBJECT: SEABROOK STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: ALTERNATIVE SOURCE TERM (TAC NO. MC1097)

Dear Mr. Warner:

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The amendment revises the Seabrook licensing basis to incorporate a full-scope application of an alternative source term methodology in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Victor Nerses, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-443
Enclosures: 1. Amendment No. 100 to License No. DPR-86
2. Safety Evaluation
cc w/encls: See next page

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<u>FPL ENERGY SEABROOK, LLC, ET AL.</u>*

<u>DOCKET NO. 50-443</u>

<u>SEABROOK STATION, UNIT NO. 1</u>

<u>AMENDMENT TO FACILITY OPERATING LICENSE</u>

Amendment No. 100
License No. NPF-86

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by FPL Energy Seabrook, LLC, et al. (the licensee), dated October 6, 2003, as supplemented by letters dated May 5, May 24, July 8, September 13, 2004, and January 13, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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*FPL Energy Seabrook, LLC (FPLE Seabrook), is authorized to act as agent for the following: Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, and Taunton Municipal Light Plant. FPLE Seabrook has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-86 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 100, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 24, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 100

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

Replace the following page of the Appendix A, Technical Specifications, with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
1-3	1-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. NPF-86

FPL ENERGY SEABROOK, LLC

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By application dated October 6, 2003, as supplemented by letters dated May 5, May 24, July 8, September 13, 2004, and January 13, 2005, FPL Energy Seabrook, LLC (FPLE or the licensee) requested changes to the Technical Specifications (TSs) for Seabrook Station, Unit No. 1 (Seabrook). The supplements dated May 5, May 24, July 8, and September 13, 2004, and January 13, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 9, 2003 (68 FR 68670).

The proposed amendment would revise the Seabrook licensing basis to incorporate a full-scope application of an alternative source term (AST) methodology in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67.

2.0 REGULATORY EVALUATION

In the past, power reactor licensees have typically used U.S. Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962, as the basis for design-basis accident (DBA) analysis source terms. The power reactor siting regulation, which contains offsite dose limits in terms of whole body and thyroid dose, 10 CFR Part 100, Section 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," makes reference to TID-14844.

In December 1999, the NRC issued 10 CFR 50.67, "Accident Source Term," which provides a mechanism for licensed power reactors to replace the traditional accident source term (TID-14844) used in their DBA analyses with an AST. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors". Section 50.67 of 10 CFR requires a licensee seeking to use an AST to apply for a license amendment and requires that the application contain an evaluation of the consequences of affected DBAs. FPLE's application of October 6, 2003, as supplemented, addresses these requirements in proposing to use the AST described in RG 1.183 as the Seabrook DBA source term used to evaluate the radiological consequences of DBAs. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC)-19, "Control Room," for a loss-of-coolant accident (LOCA), main steamline break (MSLB) accident, fuel handling accident (FHA), and control rod drop accident.

The accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large LOCA. As a result of significant core damage, fission products are available for release into the containment environment. An AST is an accident source term that is different from the accident source term used in the original design and licensing of the facility and has been approved for use under 10 CFR 50.67. Although an acceptable AST is not set forth in the regulations, RG 1.183 identifies an AST that is acceptable to the NRC staff for use at operating reactors.

This safety evaluation (SE) addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements on which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183 and GDC-19. Except where the licensee has proposed a suitable alternative, the staff used the regulatory guidance in the following documents in doing this review:

- RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants”

- RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”

- RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants”

- Standard Review Plan (SRP) Section 2.3.4, “Short-Term Diffusion Estimates for Accidental Atmospheric Releases”

- SRP Section 6.4, “Control Room Habitability Systems” (with regard to control room meteorology)

- SRP Section 15.0-1, “Radiological Consequence Analyses Using Alternative Source Term”

- SRP Section 15.6.2, “Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside the Containment”

- Branch Technical Position ETSB 11-5, “Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure.”

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the technical analyses related to the radiological consequences of DBAs that were done by FPLE in support of this proposed license amendment. Information regarding these analyses was provided in Enclosure 2 of the October 6, 2003, submittal and in the supplemental letters dated May 5, May 24, and July 8, 2004. The staff reviewed the assumptions, inputs, and methods used by FPLE to assess these impacts and did independent calculations to confirm the conservatism of the FPLE analyses. However, the findings of this SE input are based on the descriptions of the analyses and other supporting information submitted by FPLE. The staff also considered relevant information in the Seabrook Updated Final Safety Analysis Report (UFSAR) and the Seabrook TSs.

3.1 Radiation Source Term

The Seabrook reactor core contains 193 fuel assemblies containing 492 kg uranium dioxide (UO_2) with a fuel enrichment range of 1.6 to 5.0 percent by weight. The core average burnup is 45 gigawatt-days per metric ton of uranium (GWD/MTU), with a licensed peak of 62 GWD/MTU. Consistent with RG 1.183 guidance, the heat generation rate in fuel assemblies with burnups greater than 54 GWD/MTU is limited to 6.3 kilowatt per foot (kW/ft). The inventory of radionuclides in the reactor core is based on a reactor power level of 3659 megawatt thermal (MWt). This core inventory supports licensed thermal powers of up to 3587 MWt (2-percent calorimetric uncertainty). The current licensed thermal power for Seabrook is 3411 MWt. The higher power level was used in determining the core inventory so that the assessment would be applicable to future uprated conditions[1] as well as the current licensed power. The core inventory was determined using the ORIGEN-2.1 isotope generation and depletion computer code using plant-specific inputs for burnup, enrichment, and burnup rates that had been assigned on the basis of sensitivity studies. The period of irradiation was selected to be sufficiently long to allow the significant radionuclides to reach equilibrium concentrations. Radioactive decay during refueling outages was conservatively ignored. The staff finds the FPLE approach to be consistent with regulatory guidance and, therefore, acceptable.

3.2 Reactor Coolant and Secondary Plant Radiation Source Term

For DBAs in which releases occur from the secondary plant, the initial concentrations of radionuclides in the reactor coolant system (RCS) and the steam generators are assumed to be the maximum values permitted by TSs. FPLE derived the RCS and secondary system source terms from Table 11.1-1 of the Seabrook UFSAR. Since the values in this table are based on an assumption of 1 percent failed fuel, the radioiodine data were normalized to the specific activity TS limit of 1.0 μCi/gm dose equivalent I-131. The proposed definition of dose equivalent I-131 and the thyroid dose conversion factors (DCFs) of Federal Guidance Report No. 11 (FGR-11) were used in this adjustment. Non-iodine species were normalized to the TS limit of $100/\bar{E}\gamma$.

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[1]By letter dated March 17, 2004, FPLE submitted License Amendment Request LAR 04-03 requesting approval of a stretch power uprate (SPU) for Seabrook. This uprate would increase the current licensed thermal power of 3411MWt by 5.2 percent to 3587MWt. Additionally, the analyses supporting the SPU rely on the full-scope implementation of AST methodology. The application for the SPU is available in the Agencywide Documents Access and Management System (ADAMS) under accession number ML040860307.

The secondary coolant specific activity TS is 0.1 μci/gm dose equivalent I-131. Since noble gases are assumed to be released immediately, the radioiodine secondary concentrations are assumed to be 10 percent of RCS specific activity. The staff finds the FPLE approach to be consistent with regulatory guidance and staff practice.

The intent of the TSs on specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the basis of the dose analyses. Historically, licensees have calculated the dose equivalent I-131 using thyroid DCFs, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE rather than the whole body dose and thyroid dose as done previously.

While the NRC staff believes that the FGR-11 DCFs identified as “effective” should be used instead of the thyroid DCFs, the staff reviewed the licensee’s methodology for acceptability. FPLE utilized Table 11.1-1 of the Seabrook UFSAR to obtain a distribution of the I-131 - I-135 isotopes in primary coolant. This distribution was based upon 1 percent fuel defects. FPLE utilized this distribution and the inhalation thyroid DCFs from FGR-11 to calculate the activity level of isotopes I-131 - I-135 at an overall primary coolant activity level of 1μCi/g of I-131. The licensee utilized this activity level to calculate the dose consequences of MSLB and steam generator tube rupture (SGTR) accidents at 1 μCi/g and at 60 μCi/g I-131. TEDE doses were calculated using effective dose equivalent (EDE) DCFs from FGR-11. The results met the acceptance criteria of RG 1.183. As long as actual reactor coolant activity levels remain below 1 μCi/g and 60 μCi/g, when calculated using the inhalation thyroid conversion factors of FGR-11, acceptable doses would result if an MSLB or STGR accident occurred and similar conditions existed as were identified in the submittal. Therefore, the staff concluded it could accept the FPLE approach of using the thyroid DCFs but the licensee’s proposed definition of dose equivalent I-131 needed to be changed to reflect the actual manner of calculation.

The FPLE-proposed definition included the sentence, “DOSE EQUIVALENT I-131 shall be that concentration of ^{131}I (micro curie per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present.” The NRC staff believes that instead of thyroid dose, the definition should have stated TEDE dose since implementation of AST involves a TEDE dose and not a thyroid dose. In addition, the FPLE definition did not specify what thyroid DCFs from FGR-11 should be used. As proposed, it could have been the thyroid DCFs from any of the Tables for inhalation, ingestion or submersion. Since the thyroid DCFs for inhalation were the ones actually used in the analysis, that is the dose conversion which should be specified in the definition. In a December 1, 2004, request for additional information (RAI) to FPLE, the following definition of dose equivalent I-131 was proposed as reflecting the methodology actually used:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro curie per gram) which alone would produce the same thyroid TEDE dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed under Inhalation in Federal Guidance Report No. 11 (FGR-11), “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion.”

In its January 13, 2005, response to the RAI, FPLE indicated their acceptance of this proposed definition and proposed the definition as noted above. The change in definition of Dose Equivalent I-131 is discussed in more detail in Section 3.5 of this SE.

3.3 Atmospheric Relative Concentration Estimates

3.3.1 Meteorological Data

The licensee used five years of hourly onsite meteorological data collected during calendar years 1998 through 2002 to generate new atmospheric dispersion factors (X/Q values) for use in this license amendment request (LAR). Wind speed and direction were measured at approximately 10 and 60.7 meters (m) above the ground and the atmospheric stability categorization was based on temperature difference measurements between these two levels. The licensee stated that the monitoring system complies with RG 1.23, "Onsite Meteorological Programs." These data were provided for staff review in the form of hourly meteorological data files (for input into the ARCON96 atmospheric dispersion computer code) and joint frequency distributions (for input to the PAVAN atmospheric dispersion computer code). The data were used to generate control room (CR), exclusion area boundary (EAB), and low-population zone (LPZ) X/Q values for the LOCA, FHA, MSLB, SGTR, reactor coolant pump shaft seizure (locked rotor), rod cluster control assembly (RCCA) ejection, failure of small lines carrying primary coolant outside containment (letdown line break), radioactive gaseous waste system leak or failure, and radioactive liquid waste system leak or failure (release to the atmosphere) events evaluated in this LAR. The resulting atmospheric dispersion factors represent a change from those used in the current Seabrook UFSAR analyses.

The staff did a quality review of the ARCON96 hourly meteorological database using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data." Further review was performed using computer spreadsheets. Examination of the data revealed the data recovery during the five-year period was consistently in the upper 90th percentiles. With respect to atmospheric stability measurements, the frequency, length, and time of occurrence of stable and unstable atmospheric conditions appeared very good. As expected, stable and neutral conditions were consistently reported to occur at night and unstable and neutral conditions during the day. Wind speed and direction frequency distributions for each measurement channel were reasonably similar from year to year and when comparing measurements between the two heights. Year-to-year similarity in wind direction frequency at each height was particularly strong.

A comparison of joint frequency distributions of the ARCON96 data (as compiled by the staff) with the joint frequency distributions used by the licensee as input to PAVAN showed reasonably good agreement.

In summary, the NRC staff has reviewed the available information relative to the onsite meteorological measurements program and the ARCON96 and PAVAN meteorological data input files provided by the licensee. On the basis of this review, the staff concludes that these data provide an acceptable basis for estimating atmospheric dispersion estimates for DBA assessments.

3.3.2 CR Atmospheric Dispersion Factors

The licensee made numerous CR X/Q calculations using guidance from RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," and provided X/Q values for the 30 postulated source and receptor pairs that were judged to result in the limiting dose cases based upon factors such as plant layout. The X/Q values were calculated using 1998 through 2002 onsite meteorological data and the ARCON96 atmospheric computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). All releases were assumed to be ground level. The resulting CR X/Q values are presented in Table 2. The NRC staff performed a qualitative spot-check of the inputs and found them generally consistent with site configuration drawings and staff practice. Specific areas of note are as follows:

- When modeling the dose for the first 2.5 hours of the postulated MSLB, RCCA ejection, and SGTR events, the licensee assumed plume rise and reduced the ground level X/Q values calculated using ARCON96 by a factor of five, consistent with guidance in RG 1.194 for postulated releases from steam relief and atmospheric dump valves that meet the specified criteria. Among the criteria, RG 1.194 states that the time-dependent vertical velocity must exceed the 95^{th}-percentile wind speed at the release point height by a factor of at least five. The licensee estimated the 95^{th}-percentile wind speeds at the heights of the main steam safety valve and atmospheric steam dump valve to be 16.7 and 16.8 miles per hour (mph), respectively. The NRC staff made confirmatory estimates from the 1998 through 2002 meteorological data and concluded that the licensee's estimates appear reasonable. The licensee estimated the minimum effluent exit velocity to be 124.8 ft/s (85.1 mph). Thus, the ratio of this minimum effluent exit speed to the 95^{th}-percentile wind speeds is greater than a factor of five.

- As addressed in RG 1.194, the licensee also reduced X/Q values to take credit for dual intakes for postulated releases to the normal and emergency control room air intakes and for releases from the refueling water storage tank (RWST) to the diesel building intakes on the north and south sides of the diesel building. In the case of the CR air intakes, the X/Q values for the limiting intake were reduced by a factor of two since flow into the intakes is equal. The X/Q values for the postulated RWST release to the diesel intakes are averages based upon Equation 5a of RG 1.194.

In summary, the staff has reviewed the licensee's assessments of CR post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. On the basis of this review, the staff concludes that the CR X/Q values in Table 2 are acceptable for use in DBA CR dose assessments.

3.3.3 EAB/LPZ Atmospheric Dispersion Factors

The licensee calculated EAB and LPZ X/Q values using the 1998 through 2002 onsite meteorological data and the PAVAN computer code which implements the guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." The EAB and LPZ distances are approximately 914 and 2012 meters, respectively. All releases were considered to be ground level. A reactor building height of 54.9 m and a minimum cross-sectional area of 2416 m^2 were used to model building wake effects. The NRC staff qualitatively reviewed the inputs to the PAVAN computer runs and found the inputs generally consistent with site configuration drawings and staff practice.

However, the staff notes that the EAB X/Q values calculated by the licensee for intervals longer than 0-2 hours are extraneous to the dose calculations for this LAR, which are based upon the 0-2 hour X/Q value only.

In summary, the NRC staff has reviewed the licensee's assessments of EAB and LPZ post-accident dispersion conditions generated from the licensee's meteorological data and atmospheric dispersion modeling. The resulting EAB and LPZ X/Q values (except for the EAB X/Q values for intervals longer than 0-2 hours) are presented in Table 2. On the basis of this review, the staff concludes that these X/Q values are acceptable for use in DBA, EAB and LPZ dose assessments.

3.4 Accident Dose Calculations

In accordance with the guidance in RG 1.183, a licensee is not required to re-analyze all DBAs for the purpose of the application, just those affected by the proposed changes. However, on approval of this amendment, the AST and the TEDE criteria will become the licensing basis for all subsequent (except equipment qualification) radiological consequence analyses intended to show compliance with 10 CFR Part 50 requirements. This protocol is supported by staff evaluations that concluded that prior DBA analyses would remain bounding for the AST and the TEDE criteria and would not require updating. In keeping with this guidance, FPLE did an evaluation of previously analyzed DBAs to decide which, if any, were affected by the proposed amendment. FPLE re-analyzed the radiological consequences of the following, affected, DBA events:

- Loss-of-coolant accident
- Fuel handling accident
- Main steamline break
- Steam Generator Tube Rupture
- Reactor Coolant Pump Shaft Seizure
- Rod Cluster Control Assembly Ejection
- Letdown Line Break
- Radioactive Gaseous Waste System Leak or Failure
- Radioactive Liquid Waste System Leak or Failure

3.4.1 LOCA

The accident considered is a double-ended rupture of the largest pipe in the RCS. The objective of this postulated DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that the emergency core cooling system (ECCS) is not effective in preventing core damage. A LOCA is a failure of the RCS that results in the loss of reactor coolant that, if not mitigated, could result in fuel damage including a core melt. The primary coolant will blow down through the break to the containment, depressurizing the RCS and pressurizing the containment. A reactor trip occurs and the ECCS is actuated to force borated water into the reactor vessel. Containment sprays actuate to depressurize the containment. Thermodynamic analyses, done using a spectrum of RCS break sizes, show that the ECCS and other plant safety features are effective in preventing significant fuel damage. Nonetheless, the radiological consequence portion of the LOCA analysis conservatively assumes that the ECCS is not effective and that substantial fuel damage occurs. For these analyses, the failure of the largest pipe in the RCS is postulated since this represents the larger challenge to mitigating the radionuclide releases. Appendix A of RG 1.183 identifies acceptable radiological analysis assumptions for a LOCA.

Core Fission Product Release

During a LOCA, it is assumed that all of the radioactive materials dissolved or suspended in the RCS liquid will be released to the containment within 30 seconds. The gap release phase begins with the onset of fuel cladding failure at about 30 seconds and is assumed to continue for 30 minutes. As the core continues to degrade, the gap release phase ends and the early in-vessel release phase begins. This phase continues for 1.3 hours. The inventory in each release phase is assumed to be released at a constant rate over the duration of the phase and starting at the onset of the phase. The LOCA source term release fraction, timing characteristics, and radionuclide grouping are tabulated in Table 3.4-1.

Table 3.4-1
LOCA Release to Containment

Radionuclide Group	Gap Release Phase *(0.5 Hours)*	Early In-Vessel Phase *(1.3 Hours)*
Noble Gases *(Xe, Kr, Rn, He)*	0.05	0.95
Halogens *(I, Br)*	0.05	0.35
Alkaline Metals *(Cs, Rb)*	0.05	0.25
Tellurium Group *(Te, Sb, Se)*	0	0.05
Barium *(Ba, Sr)*	0	0.02
Noble Metals *(Ru, Rh, Pd, Mo, Tc, Co)*	0	0.0025
Cerium Group *(Ce, Pu, Np)*	0	0.0005
Lanthanides *(La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)*	0	0.0002

FPLE assumes that the radionuclides released from the fuel are instantaneously and homogeneously distributed throughout the containment atmosphere as they are released from the fuel. The analysis credits two mechanisms for removing released radionuclides from the containment atmosphere:

- The first mechanism is plateout by natural deposition processes. FPLE assumes that this deposition results in a removal of elemental radioiodine at a rate of 2.23 hr^{-1} in the sprayed and unsprayed regions and a removal of aerosols at a rate of 0.1 hr^{-1} in the unsprayed region only. The elemental radioiodine deposition is based on staff guidance in SRP 6.5.2; the aerosol deposition is based on the Industry Degraded Core (IDCOR) Rulemaking Program Technical Report 11.3. Regulatory Position A.3.2 references the methodology of NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," and the NRC-sponsored RADTRAD code as being acceptable to the staff. The staff compared the 0.1 hr^{-1} removal rate proposed by FPLE against the data in Table 2.2.2.1-3 of NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," and determined it to be more conservative. As such, the staff finds the value of 0.1 hr^{-1} to be acceptable.

- The second mechanism during a LOCA is removal by the containment building spray (CBS), which is automatically started by containment pressure instrumentation during a LOCA and reaches rated spray flow in 65 seconds from the event initiation. Based on evaluations of spray nozzle coverage and containment arrangement, FPLE projects that 85 percent of the containment free volume is sprayed. FPLE modeled the containment as comprising two regions–sprayed, and unsprayed, and assumes a mixing rate between the sprayed and unsprayed regions of two turnovers of the unsprayed region each hour. FPLE assumed that containment sprays were effective for particulate and elemental radioiodine, but no credit for spray removal was assumed for noble gases or for organic forms of radioiodine. The effectiveness of the sprays for radionuclide scrubbing is represented by the removal rate (often referred to as spray coefficients or spray lambda, λ). FPLE assumes an elemental radioiodine spray removal rate of 20 hr^{-1} until a decontamination factor of 200 is reached at about 2.92 hours after the event initiation, and a particulate radioiodine spray removal rate of 5.75 hr^{-1} until a decontamination factor of 50 is reached at about 3.56 hours after the event initiation, at which time the removal rate is decreased to 0.575 hr^{-1}.

FPLE assumes that the radioiodine released to the containment atmosphere consists of 95 percent cesium iodide (CsI), 4.85 percent elemental radioiodine, and 0.15 percent organic forms. This radioiodine speciation is appropriate if the containment sump pH is maintained at a value of 7.0 or higher. This is accomplished at Seabrook by chemical injection into the CBS system. Section 6.5.2.2 of the Seabrook UFSAR states that the containment sump pH will range between 8.7 and 9.2 pH during spray recirculation. The staff finds that the proposed source term assumptions are consistent with the guidance of RG 1.183 and are, therefore, acceptable.

Post-Accident Containment Sump Chemistry Management

In NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," the NRC staff concluded that iodine entering the containment from the RCS during an accident would be composed of at least 95 percent CsI, with no more than 5 percent of iodine (I) and hydriodic acid (HI). The radiation-induced conversion of iodide in water into elemental iodine (I_2) is strongly dependent on the pH. The staff stated in the NUREG that without pH control, a large fraction of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be re-evolved into the containment atmosphere if the pH is less than 7. Conversely, if the pH is maintained above 7, less than 1 percent of the dissolved iodine will be converted to elemental iodine.

The licensee described the methodology used to calculate the post-accident containment sump water pH as follows:

- within one hour of the loss-of-coolant accident (LOCA), all the borated water from the refueling water storage tank (RWST) and sodium hydroxide from the spray additive tank are added to the sump;

- sump pH will be affected only by hydrochloric acid from the irradiation of electrical cables, nitric acid from the irradiation of water and air, and sump water temperature changes;

- the formation of hydrochloric and nitric acids are based on the guidelines in NUREG/CR-5950, "Iodine Evolution and pH Control," using the 30-day integrated doses in the sump and containment atmosphere; and

- consideration of hydroiodic acid was not included since a small amount is released compared with the amount of hydrochloric and nitric acids produced.

The methodology described in NUREG/CR-5950 and NUREG-1465, and NUREG/CR-5732, "Iodine Chemical Forms in LWR [light-water reactors] Severe Accidents," can be used to determine (1) the formation of hydrochloric and nitric acids, (2) the sump pH transient, and (3) long-term iodine re-evolution. The staff has found the use of the methods in these documents acceptable for calculating post-accident sump pH for determining long-term iodine re-evolution.

The NRC staff reviewed the licensee's methodology for determining the 30-day post-accident pH value and found the evaluation included the appropriate assumptions for determining sump pH. In addition, the staff found it acceptable to assume a 30-day integrated dose for the formation of hydrochloric and nitric acids because these acid sources are time-dependent on exposure to radiation (i.e., more acids are formed with time because of increased radiation exposure).

The NRC staff independently verified through calculations that the sump pH is at least 8.6 considering the sources of borated water and sodium hydroxide. In addition, the staff determined that the production of hydrochloric and nitric acids produces results of less than a 0.1 change in pH after 30-days.

Therefore, the staff finds that the licensee's methodology for calculating the post-accident containment sump pH is acceptable because the methodology includes the appropriate assumptions consistent with the staff-approved methodologies.

<u>Iodine Removal Coefficients</u>

In SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," the effectiveness of the containment spray system can be estimated by considering the chemical and physical processes that can occur during an accident in which the system operates. The concentration of iodine in the spray solution is an important factor in determining the effectiveness of the sprays against elemental iodine vapor during an accident.

The elemental iodine and particulate spray removal coefficients represent the rate at which elemental iodine vapor and particulate fission products are removed from the containment atmosphere by the spray system. The licensee credits the maximum elemental iodine spray removal coefficient, λ_s, of 20 hr^{-1} consistent with the SRP although a higher value was calculated. In addition, the licensee calculated a particulate removal coefficient, λ_p, of 5.75 hr^{-1} and a decontamination factor (DF) of 200 for elemental iodine.

The NRC staff independently verified through calculations that the iodine removal coefficients calculated and the decontamination factor of 200 for elemental iodine will be reached within 30 days of the accident. Therefore, the staff finds that these values are acceptable for the fission product cleanup model.

Release Paths

Once dispersed in the containment, the release to the environment is assumed to occur through four pathways:

- Containment purge at event initiation
- Release from containment leakage
- Sump water leakage from ECCS systems outside of the containment
- Release from RWST due to ECCS backleakage.

Containment Purge at Event Initiation

FPLE assumes that a containment purge is in progress at the start of the LOCA; providing a path for releases to the environment. This purge is projected to be isolated within five seconds as a result of a containment isolation signal. Since the onset of radionuclide releases from the fuel occurs at 30 seconds, this pathway is isolated prior to fuel damage occurring. For purposes of analysis, FPLE assumes that the entire RCS inventory of volatile radionuclides is released to the containment from where it enters the environment at a rate of 1000 cfm for five seconds. Since fuel damage and containment sprays will not have commenced at this time, the chemical form of the radioiodine released from the RCS is assumed to be 97 percent elemental and 3 percent organic. The NRC staff finds these assumptions to be acceptable.

Containment Leakage Release

The containment structure is a reinforced concrete cylinder with a hemispherical dome and a reinforced concrete foundation. A welded steel liner is anchored to the inside face of the concrete as a leak-tight membrane. Fans maintain the pressure in the space between the containment structure and the steel liner at a value slightly below the atmospheric pressure following a LOCA. All joints and penetrations are sealed to ensure air tightness. The containment building holds up the majority of the radioactivity released from the core. FPLE assumes that the containment leaks at a rate of 0.15 percent volume per day for the first 24 hours and 0.075 percent volume per day for days 2 through 30. FPLE assumes that after eight minutes following the onset of the event, 40 percent of the containment leakage is collected by the containment enclosure and is released to the environment as a filtered ground level release. Seabrook TS 3/4.6.5 requires a draw-down time of less than four minutes. The 40-percent leakage collection is based on the acceptance criteria for Type B and Type C leakage via penetrations and isolation valves, which is 60 percent of design leakage (0.6 La). This leakage is conservatively assumed to bypass the containment enclosure and enter the environment as unfiltered ground level releases. FPLE does not assume mixing of the containment leakage in the containment enclosure.

Sump Water Leakage from ECCS Systems Outside of the Containment

During a LOCA, some radionuclides released from the fuel will be carried to the containment sump via spillage from the RCS or by transport of activity from the containment atmosphere to the sump by containment sprays and natural processes such as deposition and plateout. During the initial phases of a LOCA, safety injection and containment spray systems draw water from the RWST. Approximately 26 minutes after the start of the event these systems start to draw water from the containment sump instead. This recirculation flow causes contaminated water to be circulated through piping and components outside of the containment where small amounts of system leakage could provide a path for the release of radionuclides to the environment. FPLE assumes that the leakage rate is two times the expected value, or 48 gallons per day.

FPLE conservatively assumes that all of the radioiodines released from the fuel are instantaneously moved to the containment sump; noble gases are assumed to remain in the containment atmosphere. FPLE deviated from the guidance of RG 1.183 and proposed the following alternative treatment of the radioiodine release from the leaked ECCS liquid:

- FPLE posits that since the containment sump pH is maintained greater than 7, the radioiodine in the sump solution of nonvolatile iodide or iodate form and, as such, the chemical form of radioiodine in the sump water at the time of recirculation would be 98.85 percent aerosol, 1.0 percent elemental and 0.15 percent organic.

- FPLE assumes that all of the elemental and organic radioiodine available for release is assumed to become airborne and leak to the environment, via the plant vent, for 30 days after the start of recirculation.

RG 1.183 states that 10 percent of the total iodine entrained in the leakage flow is to be assumed to flash and enter the building atmosphere. FPLE assumptions result in only 1.15 percent of the total iodine flashing. The staff challenged the FPLE assumptions in an RAI dated December 1, 2003. In a teleconference on April 20, 2004, the staff stated their objection to the proposed response. FPLE submitted a formal response on May 24, 2004. In this response, FPLE provided a justification for their proposed alternative to the RG 1.183 guidance, and provided the results of an analysis done using the RG 1.183 assumption. The FPLE argument was not persuasive and the NRC staff has based this SE on the revised analysis that used the RG 1.183 assumptions regarding the release of radioiodine. Although FPLE arguments regarding the radioiodine speciation in the sump water have merit with regard to the containment sump, they do not address the uncertainty related to changes in the radioiodine speciation as the leaked fluid is atomized as small droplets, flows into a building sump for which the pH might be acidic, or evaporates to dryness on building surfaces. In their January 13, 2005, response to a December 1, 2004, RAI, FPLE stated, “The 10% flashing factor for Emergency Core Cooling System (ECCS) leakage included in the response to RAIs 6D1 and 6D2 will be the flashing factor utilized for the Seabrook Station licensing basis.” Therefore, this issue has been satisfactorily resolved.

<u>Release Due to ECCS Back Leakage to the RWST</u>

Although the RWST is isolated during recirculation, design leakage through ECCS valving provides a pathway for leakage of the containment sump water back to the RWST. The RWST is in the plant yard and is vented to the atmosphere. The radionuclides entrained in the back leakage could be released to the environment via the RWST vent. As such, the dose consequences are considered.

The concentrations of the radionuclides in the containment sump water are modeled as was done above for ECCS leakage. FPLE assumes that containment sump water leaks into the RWST at a rate of 0.9595 gpm starting at about 26 minutes and continuing for 30 days. The pH of the water in the RWST at the time of suction transfer is greater than 7.0 due to the sodium hydroxide added directly to the tank at the start of the event. FPLE assumes that the radioiodine speciation in the RWST would be 99 percent aerosol and 1percent elemental. FPLE states that it is their position that no elemental radioiodine would be present but, that in the interest of conservatism, they assume that 1 percent of the particulate radioiodine converts to elemental radioiodine. The staff finds the elemental radioiodine assumption acceptable for the Seabrook RWST leakage assessment since the leakage is collected in a pool of water, which has an elevated pH, and for which complete evaporation is not expected. This elemental radioiodine is assumed to become volatile and will partition between the liquid and vapor space in the RWST. FPLE used partition coefficients from NUREG/CR-5950.

The release of elemental radioiodine from the vapor space is calculated based upon the displacement of air by the incoming leakage and the expansion due to diurnal heating and cooling cycles of the tank and its contents. FPLE assumed that the tank internal temperatures would swing as high as the daily outside temperature swing of 18.2 EF, based on 2001 American Society of Heating and Air Conditioning Engineers data for the Portland, Maine, area. This is conservative in that it ignores the thermal mass of the tank and its contents. Also, the tank is in a building that will mitigate the daily temperature swing to which the tank is subjected. The analysis does not assume any heat losses from the tank either by evaporation or conduction for the 30 days of the event. This assumption establishes a conservatively high water temperature, which increases radioiodine partitioning. No flow restriction is assumed in the tank vent path and the tank is assumed to remain at atmospheric pressure. This assumption maximizes the daily release flow.

The licensee’s assumption that the particulate iodine activity in the RWST available for release was determined to be acceptable for the following reasons:

1) FPLE indicated that the sump liquid being recirculated in the ECCS will have a pH>7 and will consist of iodine in particulate form and other isotopes in particulate form.

2) The backleakage will be to the RWST outlet piping which is filled with water. Thus, there will be no flashing associated with the backleakage.

3) The backleakage will be discharged to the RWST underneath the water level of the RWST.

4) FPLE indicated that the pH of the liquid in the RWST, prior to backleakage entering the tank, is 7.1. Therefore, it is basic. The backleakage pH is also basic. Since the liquid in the RWST remains basic, there will be no dissociation of the iodine in the RWST. The iodine which will be released from the RWST vent will be that volume of air displaced by the backleakage and that volume attributed to the expansion of air in the tank due to the diurnal change in temperature of the tank.

LOCA CR Modeling

The CR normal ventilation isolation will be activated by a high containment pressure signal at event initiation. FPLE assumes an isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm. FPLE assumed an unfiltered inleakage rate of 150 cfm, of which 20 cfm is via the emergency fire exit doors and the remainder via the diesel building. This assumed inleakage rate is greater than that determined in recently performed tracer gas infiltration testing, which showed Train A inleakage to be 8±11 standard ft^3/minute (SCFM) and Train B inleakage to be 14±22 SCFM. The staff's acceptance of the unfiltered inleakage assumption does not constitute approval of the FPLE Generic Letter 2003-01 response. That response is being evaluated as a separate matter.

LOCA Summary

The staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for the LOCA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4.2 FHA

The accident considered is the dropping of a spent fuel assembly during refueling. This event could occur inside the containment or in the fuel storage building. The affected assembly is assumed to be that with the highest inventory of radionuclides of the 193 assemblies in the core. All of the fuel rods in the assembly are conservatively assumed to rupture. Volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod clad. The radionuclide inventory in the fuel rod gap of the damaged fuel rods is assumed to be instantaneously released because of the accident. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool (SFP) depending on their physical and chemical form. Appendix B of RG 1.183 identifies acceptable radiological analysis assumptions for an FHA.

FPLE assumed no decontamination for noble gases, an effective decontamination factor of 200 for radioiodines, and retention of all aerosol and particulate radionuclides. FPLE assumed that 100 percent of the radionuclides released from the reactor cavity or SFP are released to the environment in two hours without any credit for filtration, holdup, or dilution. Consistent with TS requirements, a delay of 80 hours prior to moving irradiated fuel was assumed. With the exception of different release points, the assumptions and inputs are identical for the FHA within the containment and the FHA outside the containment. To ensure that the analysis would be bounding for both release cases, FPLE did the analysis using the atmospheric dispersion factors for most limiting combination of release point and receptor.

FPLE assumes a CR isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. This isolation would be actuated by radiation levels that are greater than two times background on Geiger-Müller (GM) radiation detectors located in the ventilation intake ductwork. FPLE showed that the radiation levels due to the

DBAs would trigger isolation. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm. FPLE assumes an unfiltered inleakage rate of 300 cfm, of which 20 cfm is via the emergency fire exit doors and the remainder via the diesel building. This assumed inleakage rate is greater than the results of recently performed tracer gas infiltration tests.

The staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for the FHA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4.3 MSLB

The accident considered is the complete severance of a main steamline outside containment. The radiological consequences of a break outside containment will bound the consequences of a break inside containment. Thus, only the MSLB outside of containment is considered with regard to dose. The faulted steam generator will rapidly depressurize and release the initial contents of the steam generator to the environment. A reactor trip occurs, main steam isolation occurs, safety injection actuates, and a loss of offsite power (LOOP) occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. Appendix E of RG 1.183 identifies acceptable radiological analysis assumptions for an MSLB.

Although, FPLE states that no fuel damage is postulated to occur because of an MSLB, two radioiodine spiking cases are considered. The first assumes that a pre-incident radioiodine spike occurred just before the event and the RCS radioiodine inventory is at the maximum value (for 100 percent power) permitted by TSs. The second case assumes the event initiates a coincident radioiodine spike. Radioiodine is released from the fuel to the RCS at a rate 500 times the normal radioiodine appearance rate for eight hours.

FPLE assumes that the faulted steam generator boils dry rapidly, instantaneously releasing the entire liquid inventory and entrained radionuclides through the faulted steamline to the environment.

Leakage from the RCS to the steam generators is assumed to be the maximum value permitted by TSs. Primary-to-secondary leakage is assumed to be 500 gpd to the faulted steam generator and 940 gpd total into the three unaffected steam generators. FPLE states that this allocation of the leakage yields the most limiting doses. The primary-to-secondary leakage continues until the RCS temperature is less than 212 °F (at about 48 hours). The Seabrook surveillance procedures used to demonstrate compliance with RCS TS leakage normalize the temperature of all leakage streams to temperatures consistent with normal power operation conditions. In converting the TS maximum allowable volumetric flow to mass flow values for input to analyses, FPLE uses a density value consistent with normal power operation conditions. This is consistent with the guidance of RG 1.183 (see Appendix F, Paragraph 5.2).

The leakage in the unaffected steam generators mixes with the bulk water and is released at the assumed steaming rate. This steaming from the unaffected steam generators is assumed to continue for eight hours. FPLE determined that the tubes in the unaffected steam generators would remain covered by the bulk water. FPLE assumes that the radionuclide concentration in the unaffected steam generator is partitioned such that one percent of the radionuclides in the bulk water enters the vapor space and is released to the environment.

FPLE assumes a CR isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. This isolation would be actuated by either safety injection signals or by radiation levels greater than two times background on GM radiation detectors located in the ventilation intake ductwork. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm. FPLE assumes an unfiltered inleakage rate of 150 cfm, of which 20 cfm is via the emergency fire exit doors and the remainder via the diesel building. This assumed inleakage rate is greater than that determined in recently performed tracer gas infiltration tests.

The NRC staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for the MSLB were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4.4 SGTR

The accident considered is the complete severance of a single tube in one of the steam generators resulting in the transfer of RCS water to the ruptured steam generator. The primary-to-secondary break flow through the ruptured tube following and SGTR results in radioactive contamination of the secondary system. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. A single atmospheric steam dump valve (ASDV) is assumed to fail open providing a continuous release path. Two cases are considered:

- A single ASDV fails open when the water level reaches 33 percent in the ruptured steam generator

- A single ASDV fails open three minutes after the reactor trip.

The failed ASDV is assumed to be closed by manual operator action 20 minutes after failing open. Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR.

FPLE states that no fuel damage is postulated to occur because of an SGTR. Two radioiodine spiking cases are considered. The first assumes that a pre-incident radioiodine spike occurred just before the event and the RCS radioiodine inventory is at the maximum value (for 100-percent power) permitted by TSs. The second case assumes the event initiates a co-incident radioiodine spike. Radioiodine is released from the fuel to the RCS at a rate 335 times the normal radioiodine appearance rate for eight hours.

For the two analyzed cases, FPLE assumed primary-to-secondary break flows ranging from 1.5 to 46.2 lbm/sec, starting at event initiation and continuing for approximately 2.8 hours for

Case 1 and 2.0 hours for Case 2. FPLE assumes that a portion of the break flow flashes to vapor, rises through the bulk water, enters the steam space, and is immediately released to the environment with no mitigation or holdup. The flashing fraction ranges from 0.179 to 0.0023. The portion of the break flow that does not flash is assumed to mix with the bulk water of the steam generator. In addition to the break flow, FPLE assumes there is primary-to-secondary leakage at the maximum value permitted by TSs. Primary-to-secondary leakage is assumed to be 313 gpd into the bulk water of the ruptured steam generator and 1127 gpd total into the bulk water of the three unaffected steam generators. FPLE states that this allocation of the leakage yields the most limiting doses. The primary-to-secondary leakage continues until the RCS temperature is less than 212 °F (at about 48 hours).

The radionuclides in the bulk water are assumed to become vapor at a rate that is a function of the steaming rate for the steam generators and the partition coefficient. FPLE determined that tubes in the unaffected steam generators would remain covered by the bulk water. FPLE assumes that the radionuclide concentration in the steam generator is partitioned such that 1-percent of the radionuclides in the steam generator's bulk water enter the vapor space and are released to the environment. The partition coefficient does not apply to the flashed break flow. The steam release from the ruptured and unaffected steam generators continues until the residual heat removal (RHR) system can be used to complete the cooldown at approximately eight hours.

FPLE assumes a CR isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. This isolation would be actuated by either safety injection signals or by radiation levels greater than two times background on GM radiation detectors located in the ventilation intake ductwork. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm. FPLE assumes an unfiltered inleakage rate of 300 cfm, of which 20 cfm is via the emergency fire exit doors and the remainder via the diesel building. The total assumed inleakage rate is greater than that determined in recently performed tracer gas infiltration tests.

The NRC staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for the SGTR were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4.5 Reactor Coolant Pump Shaft Seizure

The accident considered is the instantaneous seizure of a reactor coolant pump rotor, which is synonymous with a locked rotor accident (LRA). This causes a rapid reduction in the flow through the affected RCS loop. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage. As the LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. Appendix G of RG 1.183 identifies acceptable radiological analysis assumptions for an LRA.

FPLE assumed that 10 percent of the fuel rods fail releasing the radionuclide inventory in the fuel rod gap. A radial peaking factor of 1.65 was applied. The radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage at the TS value of 500 gpd for any steam generator and 1.0 gpm for all steam generators for eight hours. FPLE assumes that this leakage mixes with the bulk water of the steam generators and that the radionuclides in the bulk water become vapor at a rate that is a function of the steaming rate for the steam generators and the partition coefficient. FPLE assumes that the radionuclide concentration in the steam generator is partitioned such that one percent of the radionuclides in the bulk water of the steam generators enters the vapor space and is released to the environment. The steam releases from the steam generators continue until the RHR system can be used to complete the cooldown at approximately eight hours.

FPLE assumes a CR isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. This isolation would be actuated by either safety injection signals or by radiation levels greater than two times background on GM radiation detectors located in the ventilation intake ductwork. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm. FPLE assumes an unfiltered inleakage rate of 150 cfm, of which 20 cfm is via the emergency fire exit doors and the remainder via the diesel building. This assumed inleakage rate is greater than that determined in recently performed tracer gas infiltration tests.

The NRC staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for the LRA were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4.6 RCCA Ejection

The accident considered is the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected due to the reactivity spike. This failure breeches the reactor pressure vessel head resulting in a LOCA to the containment. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam to the environment. The release to the environment is assumed to occur through two separate pathways:

- Release of containment atmosphere (i.e., design leakage)
- Release of RCS inventory via primary-to-secondary leakage through steam generators.

While the actual doses from an RCCA ejection would be a composite of the two pathways, an acceptable dose from each pathway, modeled as if each was the only pathway, shows that any composite dose would also be acceptable. Appendix H of RG 1.183 identifies acceptable radiological analysis assumptions for an RCCA ejection.

FPLE conservatively assumed that 15 percent of the fuel rods fail releasing the radionuclide inventory in the fuel rod gap. It is assumed that 10 percent of the core inventory of radioiodines and noble gases are in the fuel rod gap. A radial peaking factor of 1.65 was applied. In addition, localized heating is assumed to cause 0.375 percent of the fuel to melt, releasing 100 percent of the noble gases and 25 percent of the radioiodines contained in the melted fuel to the containment. For the secondary release case, 100 percent of the noble gases and 50 percent of the radioiodines contained in the melted fuel are released to the secondary.

For the containment leakage case, the radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the containment free volume. FPLE assumes that the containment leaks at a rate of 0.15 percent volume per day for the first 24 hours and 0.075 percent volume per day for days 2 through 30. FPLE assumes after eight minutes of the event initiation that 40 percent of the containment leakage is collected by the containment enclosure and is released to the environment as a filtered ground level release. All other releases from the containment are released to the environment as an unfiltered ground level release. FPLE does not assume mixing of the containment leakage in the containment enclosure. FPLE does not credit containment spray operation as a radionuclide removal mechanism. However, FPLE does assume that natural deposition processes result in a removal of elemental radioiodine at a rate of 2.23 hr^{-1} and a removal of aerosols at a rate of 0.1 hr^{-1}. The elemental radioiodine deposition is based on staff guidance in SRP 6.5.2; the aerosol deposition is based on the Industry Degraded Core (IDCOR) Rulemaking Program Technical Report 11.3. Regulatory Position A.3.2 references the methodology of NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments,” and the NRC-sponsored RADTRAD code as acceptable to the staff. The staff compared the 0.1 hr^{-1} removal rate proposed by FPLE against the data in Table 2.2.2.1-3 of NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” and determined FPLE’s to be more conservative. As such, the staff finds the value of 0.1 hr^{-1} to be acceptable.

For the secondary release case, the radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage at the technical specification value of 500 gpd for any steam generator and 1.0 gpm for all steam generators for eight hours. FPLE assumes that this leakage mixes with the bulk water of the steam generators and that the radionuclides in the bulk water become vapor at a rate that is a function of the steaming rate for the steam generators and the partition coefficient. FPLE conservatively assumed that the chemical form of the radioiodine released to the environment would be 97 percent elemental and 3 percent organic. The FPLE assumes that the aerosol and iodine radionuclide concentration in the steam generator is partitioned such that the one percent of the radionuclides that enter the unaffected steam generators from the RCS enter the vapor space and are released to the environment. The steam releases from the steam generators continue until the RHR system can be used to complete the cooldown at approximately eight hours.

FPLE assumes a CR isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. This isolation would be actuated by either safety injection signals or by radiation levels greater than two times background on GM radiation detectors located in the ventilation intake ductwork. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm for the secondary case. For the secondary side release analysis, FPLE assumed an unfiltered inleakage rate of

150 cfm, of which 20 cfm is via the emergency fire exit doors and the remainder via the diesel building. For the containment release path, FPLE assumes an unfiltered inleakage rate of 190 cfm, of which 20 cfm is via the emergency fire exit doors and the remainder via the diesel building. These assumed inleakage rates are greater than that determined in recently performed tracer gas infiltration tests.

The staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for both cases of the RCCA ejection were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4.7 Letdown Line Rupture (LLB)

The accident considered is a failure, outside of containment, of a small line connected to the RCS pressure boundary. Such lines are used as instrument sensing lines or for RCS cleanup systems. The Seabrook licensing basis considers a double-ended rupture of the letdown line outside of containment in the primary auxiliary building. Operators take manual actions in accordance with procedures to isolate the rupture within 30 minutes, ending the release. FPLE also assumes a release from the secondary plant via the main condenser. This event is not specifically addressed in RG 1.183. Analysis guidance is provided in SRP 15.6.2.

No fuel damage is postulated to occur because of an LLB. Two radioiodine spiking cases are considered. The first assumes that a pre-incident radioiodine spike occurred just before the event and the RCS radioiodine inventory is at the maximum value (for 100 percent power) permitted by TSs. The second case assumes the event initiates a co-incident radioiodine spike. Radioiodine is released from the fuel to the RCS at a rate 500 times the normal radioiodine appearance rate for eight hours.

FPLE models the LLB flow as 140 gpm at a specific volume of 62 lbm/ft^3 for 30 minutes, with a flashing fraction of 0.1815. All of the noble gases entrained in the rupture flow and the non-noble gas radionuclides entrained in the flashed vapor are released to the environment without holdup or mitigation.

Since there is no reactor trip projected, FPLE assumes that the plant remains at power and that the main condenser is available for the 30-day duration of the event. In the submittal, FPLE stated that they were conservatively assuming that radioiodine partitioning was not applicable to steam generators at power. Instead, FPLE assumed a decontamination factor of 100 for radioiodines and aerosols in the main condenser. FPLE stated in the submittal that this treatment of releases from the main condenser was consistent with the pre-trip treatment of secondary side steam release during an SGTR.

The staff reviewed the UFSAR description for the SGTR and found no discussion of pre-trip release treatment. In particular, UFSAR Table 15.6-6 Item II.C, “Iodine Partitioning for the Main Steam Condenser” has the notation of “N/A.” The NRC staff notes that the Seabrook UFSAR analysis for the LLB does not address releases via the secondary system. Regulatory Position 5.1.2 of RG 1.183 establishes prerequisites that must be met before credit may be taken for accident mitigation features. Since the main condenser does not meet these prerequisites and since the Seabrook current licensing basis does not credit the main condenser as a release mitigation feature for an LLB, the staff found this assumption to be unacceptable. Consequently, the NRC staff requested information in their December 1, 2004, letter which addressed the release via the condenser for the letdown line rupture. In response to the release via the condenser, FPLE's January 13, 2005, response stated, "License Amendment Request (LAR) 03-02, Licensing Technical Report (page 42 of 94), Item 9, 'Regulatory Position 5.5.4 of Appendix E' states that an iodine decontamination factor of 99% will be assigned for the releases from the condenser. The 99% iodine decontamination factor occurs entirely in the steam generator. There is no decontamination assumed to occur in the condensers. Therefore, there is no difference in the iodine decontamination factor for a release from the steam generators or a release from the condensers." With this response, the release via the condenser is not an issue since no credit for removal via the condenser is assumed. It should be noted that the iodine decontamination factor utilized was actually 100 and not 99 percent as stated in the FPLE submittal. This was a typographical error identified by FPLE.

In addition to question number 6 of the December 1, 2004 letter, the staff also requested additional information concerning the FPLE's assumption of no reactor trip and no LOOP for this event. This was contained in question numbers 4 and 5 in the December 1, 2004, letter. In response to those requests, FPLE stated, "A reactor trip is not assumed to occur in the rupture of a letdown line since analyzing the event with no reactor trip maximizes the release. Thus, the event with no reactor trip is more limiting and bounds a letdown line rupture with a reactor trip event. A loss of offsite power is not assumed to occur since a loss of offsite power would result in a reactor trip. As stated in FPL Energy Seabrook's response to question 4, a reactor trip is not assumed to occur in the rupture of a letdown line event since analyzing the event with no reactor trip maximizes the release." Since FPLE has maximized the releases by assuming no LOOP and no reactor trip, the incorporation of these assumptions into the letdown line rupture event is acceptable.

FPLE assumes a CR isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. This isolation would be actuated by either safety injection signals or by radiation levels greater than two times background on GM radiation detectors located in the ventilation intake ductwork. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm. FPLE assumes an unfiltered inleakage rate of 300 cfm, of which 20 cfm is via the emergency fire exit doors and the remainder via the diesel building. This assumed inleakage rate is greater than that determined in recently performed tracer gas infiltration tests.

The staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for the LLB were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4.8 Radioactive Gaseous Waste System Leak or Failure (GWL)

The accident considered is a rupture of the gaseous waste system that releases the entire inventory of the gaseous waste delay beds to the environment with no hold-up, dilution, or filtration. The gaseous waste system collects non-condensable gases from the RCS letdown degasifiers and other sources, processes the gas, and releases it to the environment in a controlled manner. The delay beds are five large vessels filled with charcoal media that holdup radioactive gases for decay. The Seabrook UFSAR states that each bed holds about 20 ft^3 of gas. This event is not specifically addressed in RG 1.183, in an SRP chapter or a RG. The staff relied upon Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," which is contained in SRP 11.3, "Gaseous Waste Management Systems," and Seabrook's current licensing basis for this review.

FPLE assumes that the delay beds contain the design inventories of radioactive gases that are based on long-term plant operation with one percent failed fuel. FPLE assumes that the entire inventory is released at an arbitrary, but conservative, rate of 10,000 ft^3 in two hours.

FPLE assumes a control room isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. This isolation would be actuated by radiation levels greater than two times background on GM radiation detectors located in the ventilation intake ductwork. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm. FPLE assumes an unfiltered inleakage rate of 300 cfm. This assumed inleakage rate is greater than that determined in recently performed tracer gas infiltration tests.

Since RG 1.183 does not address this event, FPLE proposed a dose criterion of a small fraction of the 10 CFR 50.67 criteria, or 2.5 rem TEDE. The Seabrook UFSAR discussion for this event concluded with the statement that the doses are below the values specified in 10 CFR Part 100. FPLE proposed using the "small fraction" modifier on the basis that the UFSAR criteria for the liquid waste release event used this criterion.

As noted in Branch Technical Position 11-5, the acceptance criterion used for consequences of the release of the contents of the offgas system from a pressurized-water reactor (PWR) is limited to Part 20 for facilities with waste gas decay tanks. However, for PWRs with charcoal beds as the processing mechanism for waste gas decay tanks, Section 5.6.1 of NUREG-0133, issued October 1978, specifically calls out the limit as being a small fraction of 10 CFR Part 100 provided that the gross radioactivity measured prior to entering the adsorption system is limited by a release rate alarm setpoint with indication in the main control room. It further states in NUREG-0133 that this monitor provides reasonable assurance that the potential consequence of an accident does not result in a total body dose that exceeds a small fraction of 10 CFR Part 100.

In the December 1, 2004, RAI, the NRC staff made an inquiry as to whether Seabrook had such a release rate alarm setpoint. In addition, the staff also asked what criterion was in the Seabrook Radiological Effluent Technical Specifications for a release from this pathway. In response to this RAI, FPLE provided the following in their January 13, 2005, response: "Seabrook Station has a release rate monitor that provides indication to the Control Room. There are three monitors associated with the carbon delay beds: (1) a monitor upstream of the carbon delay beds that provides indication and alarm; (2) a monitor that indicates the degradation of the absorption properties of the carbon delay beds that provides indication and alarm; and (3) a monitor downstream of the carbon delay beds that provides indication, alarm and isolation. The downstream monitor also has the capability of maintaining a running inventory of the total activity vented to the atmosphere. The criterion for a release from this pathway is based on 10 CFR Part 20. As stated in the Seabrook Station Offsite Dose Calculation Manual (ODCM), the alarm/trip setpoints for the radioactive gaseous effluent instrumentation are calculated to ensure that the alarm and trip will occur prior to exceeding the limits of 10 CFR Part 20."

Based upon the Seabrook design for the processing of the waste gas and the above response, the staff finds the acceptance criterion proposed for Seabrook acceptable.

The staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for the GWL were found to meet the applicable accident dose criteria, as discussed above, and are, therefore, acceptable.

3.4.9 Radioactive Liquid Waste System Leak or Failure (LWL)

The accident considered is the unexpected and uncontrolled release of fission gases from radioactive liquids contained in the liquid waste system. The liquid waste system collects contaminated liquids from various plant systems including the chemical and volume control system and boron recovery system, processes the liquid, and releases it to the environment in a controlled manner. These two systems contain significant inventories of fission gases. This event is not specifically addressed in RG 1.183, in an SRP chapter or in a RG. Thus, The NRC staff relied upon Seabrook's current licensing basis for this review.

FPLE assumes that the radioactive inventory of noble gases and iodines in the boron waste storage tank and letdown degasifier is shown in UFSAR Table 15.7-8, based on one percent failed fuel. FPLE assumes that the entire gaseous inventory is released at an arbitrary, but conservative, rate of 10,000 ft^3 in two hours.

FPLE assumes a control room isolation delay of 30 seconds to account for diesel generator sequencing, damper positioning, and instrumentation delays. This isolation would be actuated by radiation levels greater than two times background on GM radiation detectors located in the ventilation intake ductwork. Following isolation, the filtered outside air makeup is 600 cfm, and the filtered recirculation flow is 390 cfm. FPLE assumes an unfiltered inleakage rate of 300 cfm. This assumed inleakage rate is greater than the results of recently performed tracer gas infiltration tests.

The Seabrook UFSAR discussion for this event concluded with the statement that the doses are a small fraction of the values specified in 10 CFR Part 100. Since RG 1.183 does not address this event, FPLE proposed to continue to apply the current licensing basis "small fraction" modifier to the 10 CFR 50.67 criteria to arrive at an acceptance criterion of 2.5 rem TEDE. The staff believes that a criterion of 100 mrem TEDE (as derived from Branch Technical Position ETSB 11-5, updated to reflect the revised 10 CFR Part 20), should be applicable to this event. In response to the NRC staff's December 1, 2004, RAI concerning the criterion for this event, FPLE provided the following:

> Seabrook Station UFSAR Section 15.7.2, "Radioactive Liquid Waste System (RLWS) Leak or Failure (Release to Atmosphere)," evaluates the radiological consequence of a release to the atmosphere of radioactive fission gases from an unexpected and uncontrolled release of radioactive liquids contained in waste systems. This event

analyzes atmospheric releases from the rupture of either the boron waster storage tank or a letdown degasifier. The Radioactive Liquid Waste System Failure was re-analyzed using Alternate Source Term Methodology to remain consistent with the UFSAR Chapter 15 events.

Regulatory Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors,” does not provide any requirement or dose limits for a RLWS failure; therefore, the acceptance criteria were set by the current Seabrook Licensing basis. Section 15.7.2.4 of the current Seabrook UFSAR concludes only that the consequences are within a “small fraction” of the values specified in 10 CFR Part 100. Therefore, the off-site dose acceptance criteria were established as 10 percent of the 10 CFR 50.67 limits.

Upon further review, FPL Energy Seabrook concurs with the NRC’s assessment that 10 CFR Part 20 limits are more appropriate for an RLWS event. Therefore, the acceptance criteria for the radioactive liquid waste system failure is changed to 100 mrem TEDE. Actual analysis for this event indicates dose will be below this limit.

Thus, the NRC staff finds this change in acceptance criterion for this event acceptable.

The staff found that FPLE used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE. The assumptions found acceptable to the staff are presented in Table 1. Additionally, the NRC staff did independent calculations and confirmed the FPLE conclusions. The EAB, LPZ, and CR doses estimated by FPLE for the LWL were found to meet the applicable accident dose criteria, as discussed above, and are, therefore, acceptable.

3.5 TS 1.1, “Definitions,” Change

This proposed change would revise the definition of dose-equivalent I-131 in TS Section 1.10 to replace the current reference to RG 1.109 with the proposed reference to FGR-11.

The intent of the TSs on specific activity is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the basis of the dose analyses. Historically, licensees have calculated the dose equivalent I-131 using thyroid DCFs, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE rather than the whole body dose and thyroid dose as done previously.

While the NRC staff believes that the FGR-11 DCFs identified as “effective” should be used instead of the thyroid DCFs, the staff reviewed the licensee’s methodology as to its acceptability. FPLE utilized Table 11.1-1 of the Seabrook UFSAR to obtain a distribution of the ^{131}I - ^{135}I isotopes in primary coolant. This distribution was based upon one-percent fuel defects. FPLE utilized this distribution and the inhalation thyroid DCFs from FGR-11 to calculate the activity level of isotopes ^{131}I - ^{135}I at an overall primary coolant activity level of 1 μCi/g dose equivalent ^{131}I. The licensee utilized this activity level to calculate the dose consequences of MSLB and SGTR accidents at 1 μCi/g and at 60 μCi/g dose equivalent ^{131}I. TEDE doses were calculated using EDE DCFs from FGR-11. The results met the acceptance criteria of RG 1.183. As long as actual reactor coolant activity levels remain below 1 μCi/g and 60 μCi/g, respectively, when calculated using the inhalation thyroid conversion factors of FGR-11, acceptable doses would result if an MSLB or STGR accident occurred and similar conditions existed as were identified in the submittal. Therefore, the NRC staff could accept the FPLE approach using the thyroid DCFs but the licensee's proposed definition of dose equivalent ^{131}I needed to be changed to reflect the actual manner of calculation.

The FPLE-proposed definition included the sentence, "DOSE EQUIVALENT I-131 shall be that concentration of I-131 (in micro curies per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present." This sentence was inaccurate. Instead of thyroid dose, the definition should have stated TEDE dose since implementation of AST involves a TEDE dose and not a thyroid dose. In addition, the FPLE definition did not specify what thyroid DCFs from FGR-11 should be used. As proposed, it could have been the thyroid DCFs from any of the tables for inhalation, ingestion or submersion. Since the thyroid DCFs for inhalation were the ones actually used in the analysis, that is the dose conversion which should be specified in the definition. Therefore, in a December 1, 2004, RAI to FPLE, the following definition of dose equivalent I-131 was proposed as reflecting the methodology actually used:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (micro curie per gram) which alone would produce the same thyroid TEDE dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed under Inhalation in Federal Guidance Report No. 11 (FGR-11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

In FPLE's January 13, 2005, response to the RAI, they indicated their acceptance of this proposed definition and proposed the definition as noted above. Thus, the NRC staff finds FPLE's proposed definition of January 13, 2005, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Hampshire and Massachusetts State officials were notified of the proposed issuance of the amendment. The State officials had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 68670). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by FPLE to assess the radiological impacts of the proposed full implementation of an AST at Seabrook. The staff finds that FPLE used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The staff compared the doses estimated by FPLE to the applicable criteria identified in Section 2.0. The staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The staff finds reasonable assurance that Seabrook, as modified by this amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the proposed license amendment is acceptable with regard to the radiological consequences of postulated DBA's.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the Seabrook design basis is superseded by the AST proposed by FPLE. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR Part 50.67 or fractions thereof, as defined in Regulatory Position 4.4 of RG 1.183. All future radiological accident analyses done to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined in the Seabrook design basis, and modified by the present amendment.

Since these analyses were done at a power level of 3659 MWt (102 percent of 3587 MWt), the staff finds that the radiological consequences of the DBAs would remain bounding up to a licensed thermal power of 3587 MWt. However, the approval of this amendment does not confer authority to operate above the current licensed rated thermal power.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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TABLE 1

ANALYSIS ASSUMPTIONS

Assumptions Common to One or More Analyses

Reactor power level, MWt (includes uncertainty)		3659
Initial RCS activity (1.0μCi/gm dose equivalent I-131)		Submittal Table 1.7.2-1
Initial secondary activity (0.1 μCi/gm dose equivalent I-131)		Submittal Table 1.7.3-1
Core fission product inventory		Submittal Table 1.7.4-1
Dose conversion factors		FGR 11 & 12
Offsite breathing rate, m^3/sec.		
0-8 hours		3.47E-4
8-24 hours		1.75E-4
24-720 hours		2.32E-4
Control room volume, ft^3		246,000
Control Room HVAC system	Normal	Emerg.
Filtered air makeup, cfm	0	600
Unfiltered air makeup, cfm	1000	0
Recirculation, cfm	0	390
Unfiltered inleakage, cfm	varies by accident	
Intake filter efficiency,%		
Aerosols	99	99
Elemental/organic	95	95
Control room breathing rate, m^3/sec.		3.47E-4
Control room occupancy factors		
0-24 hours		1.0
1-4 days		0.6
4-30 days		0.4
Offsite χ/Q, sec/m^3		
EAB: 0-2 hr		3.17E-4
LPZ: 0-2 hr		1.54E-4
0-8 hr		8.63E-5
8-24 hr		6.46E-5
24-96 hr		3.45E-5
96-720 hr		1.40E-5
Control room χ/Q values		Table 2

Assumptions for LOCA Analysis

Onset of gap release phase, sec.	30

Core release fractions and timing–containment atmosphere

Duration, hrs	0.5000E+00	0.1300E+01
Noble Gases:	0.5000E-01	0.9500E+00
Iodine:	0.5000E-01	0.3500E+00
Cesium:	0.5000E-01	0.2500E+00
Tellurium:	0.0000E+00	0.5000E-01
Sr, Ba:	0.0000E+00	0.2000E-01
Noble Metals:	0.0000E+00	0.2500E-02
Cerium:	0.0000E+00	0.5000E-03
Lanthanum:	0.0000E+00	0.2000E-03

Containment iodine species distribution	
Elemental	0.95
Organic	0.0485
Particulate	0.0015
Control room isolation and switchover to emergency mode, sec.	30
Control room unfiltered inleakage	
via diesel building, cfm	130
via fire door, cfm	20

Containment Leakage Pathway

Containment free volume, ft^3	2.704E6
Sprayed fraction	0.85
Sprayed, ft^3	2.309E6
Unsprayed, ft^3	3.95E5
Containment release	
0-24 hours, %/day	0.15
24-720 hours, %/day	0.075
Containment iodine removal	
containment sprayed fraction	0.854
Spray start, sec.	65
Sprayed/unsprayed mixing rate, unsprayed volume/hour (cfm)	2 (13000)
Maximum iodine DF - Elemental (Particulate)	200 (50)
Time to reach DF, hours	
Elemental	2.92
Particulate	3.56
Elemental iodine removal rate, 1/hr	20
Particulate iodine removal rate, 1/hr	
Prior to DF = 50	5.75
After DF is reached	0.58
Elemental iodine removal by wall deposition, 1/hr	2.23
Particulate iodine removal by natural deposition, 1/hr (unsprayed region only)	0.1

Secondary containment drawdown time, min	4.5
Secondary containment filtration	
Aerosols/Elemental	95
Organic	85
Secondary containment bypass fraction	0.6
Release points	
Leakage	Plant vent
Secondary containment bypass	containment surface

ECCS Leakage Pathway

Start of ECCS leakage, minutes	26
ECCS leak rate (includes 2x multiplier), gpd	48
Duration of release, days	30
Containment sump volume, ft^3	69,159
Iodine flashing (fraction of total iodine in leakage)	0.1
Fraction of core inventory iodine in sump	0.4
Chemical form release fractions	
Elemental	0.97
Organic	0.03
Release pathway	Plant vent

<u>Release from RWST</u>

Containment sump water backleakage to RWST (includes 2x multiplier), gpm	0.9595
Initial RWST liquid inventory at time of recirculation, gal	47,000
Diurnal temperature swing, EF	18.2

Iodine release rate (applied to containment sump inventory)

Hours	Rate, cfm
0	1.04E-5
22	2.72E-5
24	6.48E-5
100	1.02E-4
200	1.31E-4
300	1.53E-4
400	1.70E-4
500	1.84E-4
600	1.85E-4
700	1.80E-4

RWST chemical form release fractions	
Elemental	0.01
Aerosol	0.99
Release point	RWST

Release from Containment Purge

Source term	Submittal Table 1.7.2-1
Release rate, cfm	1000
Release duration, sec.	5

Assumptions for FHA Analysis

Radial peaking factor	1.65
Number of fuel assemblies in core	193
Number of fuel assemblies damaged	1
Delay before spent fuel movement, hrs	80
Source term	Submittal Table 1.7.5-1
Iodine decontamination factor	
Elemental	285
Organic	1
Chemical form of iodine in pool, fraction	
Elemental	0.9985
Organic	0.0015
Release duration, hrs	2
Release filtration or holdup	None credited
Control room isolation and switchover to emergency mode, sec.	30
Control room unfiltered inleakage	
via diesel building, cfm	280
via fire door, cfm	20
Release point	Containment personnel hatch

Assumptions for MSLB Analysis

Pre-incident iodine spike activity (60 µCi/gm dose equivalent I-131)	Submittal Table 2.3-3
Co-incident spike appearance rate, based on	Submittal Table 2.3-5
RCS letdown flow rate (115F, 2235 psia), gpm	132.0
RCS letdown demineralizer efficiency	4
RCS mass, lbm	505,000
RCS leakage, gpm	11
Co-incident spike multiplier	500
Iodine spike duration, hrs	8
Chemical form release fractions	
Elemental	0.97
Organic	0.03
Primary-to-secondary leakage	
Faulted steam generator (SG), gpd	500
To three unaffected SGs, gpd	940
Duration, hours	48
Release duration, hrs	
Faulted SG	48
Unaffected SGs	8
Liquid Masses, lbm	
RCS	539,037
Faulted S/G	166,000
All intact S/G	297,912
Steam release from faulted SG	
Instantaneous, lbm	166,000
0-8 hr, gpd	500
Steam release from all unaffected SGs, lbm/min	
0-2 hours	3383
2-8 hours	2564
Steam partition coefficient in SGs	
Faulted SG	1.0
Unaffected SG	0.01
Control room isolation and switchover delay, sec	30
Control room infiltration	
to CR fire exit, cfm	20
to Diesel bldg, cfm	130
Release points	Closest main steamline Closest MSSV

Assumptions for SGTR Analysis

Pre-incident iodine spike activity (60 µCi/gm dose equivalent I-131)	Submittal Table 2.4-3
Co-incident spike appearance rate, based on	Submittal Table 2.4-5
RCS letdown flow rate (115F, 2235 psia), gpm	132.0
RCS letdown demineralizer efficiency	4
RCS mass, lbm	505,000
RCS leakage, gpm	11
Co-incident spike multiplier	335
Iodine spike duration, hrs	8
Release duration, hrs	
Ruptured SG	8
Unaffected SGs	8
Liquid Masses, lbm	
RCS	539,037
Ruptured S/G	99,304

Tube break flow information

ADV Failure Case 1			ADV Failure Case 2		
Time Hr	Break Flow lbm/s	Flash Fraction	Time Hr	Break Flow lbm/sec	Flash Fraction
0.0	12.5	0.177	0	12.5	0.177
0.00278	46.2	0.179	0.00278	46.2	0.179
0.274	34.9	0.072	0.274	38.1	0.113
0.5	36.7	0.061	0.417	43.1	0.148
0.754	42.7	0.124	0.555	43.3	0.138
1.0	43.8	0.115	0.694	43.8	0.128
1.25	41.4	0.040	0.825	40.1	0.134
1.46	37.2	0.0023	1.03	36.6	0.0548
1.71	37.3	0.0	1.20	37.5	0.0142
1.76	34.1	0.0	1.38	39.0	0.0
1.78	26.2	0.0	1.43	17.3	0.0
1.79	3.9	0.0	1.50	8.3	0.0
1.83	4.6	0.0	1.78	2.9	0.0
1.90	12.7	0.0	1.89	1.5	0.0
2.0	12.6	0.0	2.00	0.0	0.0
2.78	0.0	0.0	0.0	0.0	0.0

Steam generator release data

ADV Failure Case 1			ADV Failure Case 2		
Time Hr	Unaffected lbm/min	Ruptured lbm/min	Time Hr	Unaffected lbm/min	Ruptured lbm/min
0.0	217,542	72,393	0.0	217,542	72,393
0.00278	216,967	73,140	0.00278	216,967	73,140
0.274	3,630	2,743	0.274	4,752	7,782
0.5	3,630	11,860	0.417	4,752	6,446
0.754	3,630	7,032	0.555	4,752	5,547
1.0	3,630	4,843	0.694	4,752	4,819
1.25	3,630	13.9	0.825	2,361	0
1.46	9,959	0	1.03	15,738	0
1.78	1,934	0	1.20	4,393	0
2.0	3,056	42.6	1.89	4,772	0
8.0	0	0	2.0	3,056	42.6
720.0	0	0	8.0	0	0
			720.0	0	0

Primary-to-secondary leakage	
Ruptured SG, gpd	313
To three unaffected SGs, gpd	1127
Duration, hours	48
Chemical form release fractions	
Elemental	0.97
Organic	0.03
Steam partition coefficient in SGs	
Ruptured SG flashed flow	1.0
Ruptured SG non-flashed flow	0.01
Unaffected SG	0.01
Release point	
<2.5 hours	Closest MSSV
>2.5 hours	Closest ARV
Control room isolation and switchover delay, sec	30
Control room unfiltered infiltration	
from CR fire exit, cfm	20
from Diesel bldg, cfm	280

Assumptions for LRA Analyses

Radial peaking factor		1.65
Fraction of fuel that exceeds DNB		0.10
Fraction of core inventory in gap		
Kr-85		0.10
I-131		0.08
Alkali metals		0.12
Other noble gases / iodines		0.1
Iodine speciation	CNMT	Secondary
Aerosol	0.95	0
Elemental	0.0485	0.97
Organic	0.0015	0.3
RCS mass, lbm		
Minimum (for fuel failure dose contribution)		434,044
Maximum (for RCS initial dose contribution		539,037
Primary to secondary leakage, gpm		1.0
Primary to secondary leakage duration, hours		8
Steam generator mass, @ lbm/SG		99,304
Steam partition coefficient in SGs		0.01
Steam release rate from SGs, lbm/min		
0-2 hours		3392
2-8 hours		2675
Control room isolation and switchover delay, sec		30
Control room unfiltered infiltration		
from CR fire exit, cfm		20
from diesel bldg, cfm		130
Release point		
<2.5 hours		Closest MSSV
>2.5 hours		Closest ARV

<u>Assumptions for Control Rod Ejection Accident Analyses</u>

Radial peaking factor		1.65
Fraction of rods that exceed DNB		0.15
Gap fraction, all nuclide groups		0.10
Fraction of rods in core that experience melt		0.00375
DNB isotopic composition for noble gases and iodine		0.10
Melt isotopic composition	CNMT	SG
Noble gases	1.0	1.0
Iodine	0.25	0.5
Iodine species fraction	CNMT	SG
Particulate/aerosol	0.95	0
Elemental	0.0485	0.97
Organic	0.0015	0.03
Containment free volume, ft^3		2.704E6
Containment Sprays		Not credited
Containment release		
0-24 hours, %/day		0.15
24-720 hours, %/day		0.075
Containment natural deposition (elemental) 1/hr		2.2
Containment particulate deposition 1/hr		0.1
Duration of release, days		30
Secondary containment drawdown time, sec.		480
Secondary containment filtration		
Aerosols/elemental		95
Elemental/organic		85
Secondary containment bypass fraction		0.6
RCS mass, lbm		
Minimum (for fuel failure dose contribution)		434,044
Maximum (For RCS initial dose contribution)		539,037
Primary to secondary leakage, gpm		1.0
Primary to secondary leakage duration, hours		8
Steam generator mass@, lbm/SG		99,304
Steam partition coefficient in SGs		0.01
Steam release rate from SGs, lbm/min		
0-2 hours		3392
2-8 hours		2675
Control room isolation and switchover delay, sec.		30

Control room unfiltered inleakage, cfm	
Containment	190
Secondary	150
Release points	
Containment leakage	Plant vent
Containment bypass	Containment surface
Secondary	Closest MSSV/SRV

<u>Letdown Line Rupture</u>

RCS mass, lbm	
Minimum (for iodine spike dose contribution)	434,044
Maximum (For RCS initial dose contribution)	539,037
Pre-incident iodine spike activity (60 µCi/gm dose equivalent I-131)	Submittal Table 2.4-3
Co-incident spike appearance rate, based on	Submittal Table 2.7-2
RCS letdown flow rate (115F, 2235 psia), gpm	132.0
RCS letdown demineralizer efficiency, %	100
RCS mass, lbm	505,000
RCS leakage, gpm	11
Co-incident spike multiplier	500
Iodine spike duration, hrs	8
Control room unfiltered infiltration	
from CR fire exit, cfm	20
from Diesel bldg, cfm	280
Letdown line rupture	
Flow rate, lb/min (gpm)	1160 (140)
Flashing fraction (380F / 2235 psia)	0.1815
Duration, min	30
Filtration	None credited
Release point	Auxiliary bldg louvers

<u>Waste System Failure</u>

Release inventory	
Gaseous	Submittal Table 2.8-2
Liquid	Submittal Table 2.10-2, 2.10-3
RGWS / RLWS component volume, ft^3	10,000
Tank release assumption	Entire inventory in 2 hours
Control room isolation and switchover delay, sec	30
Control room unfiltered infiltration, cfm	300

Table 2

Seabrook Relative Concentration (X/Q) Values

Time (hr)	Receptor Location	X/Q (sec/m^3)
0 - 2 hours	Exclusion Area Boundary	3.17E-04
0 - 8 hours	Low Population Zone	8.63E-05
8 - 24 hours	Low Population Zone	6.46E-05
1 - 4 days	Low Population Zone	3.45E-05
4 - 30 days	Low Population Zone	1.40E-05

Control Room X/Q Values

Release Point	Receptor Point	0-2 hr X/Q	2-8 hr X/Q	8-24 hr X/Q	1-4 days X/Q	4-30 days X/Q
Plant Vent	East Intake	2.34 E-04	1.85 E-04	6.75 E-05	4.62 E-05	3.87 E-05
Plant Vent	CR Fire exit Door	7.54 E-04	5.03 E-04	2.00 E-04	1.45 E-04	9.89 E-05
Plant Vent	Diesel Building Intake	7.01 E-04	4.74 E-04	1.89 E-04	1.37 E-04	8.97 E-05
Closest Containment Surface Point	East Intake	4.40 E-04	3.46 E-04	1.29 E-04	8.40 E-05	6.80 E-05
Closest Containment Surface Point	CR Fire Exit Door	3.08 E-03	2.17 E-03	8.48 E-04	6.31 E-04	4.64 E-04
Closest Containment Surface Point	Diesel Building Intake	2.06 E-03	1.48 E-03	5.79 E-04	4.29 E-04	3.11 E-04
RWST	West Intake	3.54 E-04	2.75 E-04	9.70 E-05	6.90 E-05	4.37 E-05
RWST	CR Fire Exit Door	7.52 E-03	3.85 E-03	1.26 E-03	9.29 E-04	7.23 E-04
RWST	Diesel Building Intake	5.06 E-03	2.85 E-03	9.00 E-04	7.17 E-04	6.17 E-04
Containment Personnel Hatch	East Intake	2.84 E-04	2.48 E-04	1.04 E-04	6.50 E-05	5.10 E-05
Containment Personnel Hatch	CR Fire Exit Door	2.84 E-03	2.30 E-03	8.67 E-04	5.87 E-04	3.70 E-04
Containment Personnel Hatch	Diesel Building Intake	1.97 E-03	1.60 E-03	5.99 E-04	4.04 E-04	2.58 E-04
Main Steam Line Closest Point	East Intake	8.70 E-04	7.85 E-04	3.22 E-04	2.02 E-04	1.61 E-04
Main Steam Line Chase (West) Panel (North)	CR Fire Exit Door	4.55 E-03	3.72 E-03	1.38 E-03	9.67 E-04	6.35 E-04

Release Point	Receptor Point	0-2 hr X/Q	2-8 hr X/Q	8-24 hr X/Q	1-4 days X/Q	4-30 days X/Q
Main Steam Line Chase (West) Panel (North)	Diesel Building Intake	3.11 E-03	2.50 E-03	9.37 E-04	6.53 E-04	4.29 E-04
Primary Auxiliary Building Louver PAH-L6D	West Intake	3.21 E-04	2.68 E-04	1.02 E-04	6.75 E-05	3.72 E-05
Primary Auxiliary Building Fan PAH-FN46A	CR Fire Exit Door	2.91 E-03	1.98 E-03	6.61 E-04	5.09 E-04	4.37 E-04
Primary Auxiliary Building Fan PAH-FN46A	Diesel Building Intake	2.63 E-03	1.81 E-03	6.48 E-04	4.86 E-04	3.95 E-04
Turbine Building Closest Point	East Intake	8.40 E-04	7.65 E-04	3.44 E-04	2.41 E-04	1.91 E-04
Turbine Building Closest Point	CR Fire Exit Door	4.49 E-03	3.22 E-03	1.19 E-03	8.27 E-04	5.99 E-04
Turbine Building Closest Point	Diesel Building Intake	5.95 E-03	4.80 E-03	1.79 E-03	1.24 E-03	8.00 E-04
Waste Process Building SW Corner Roll-Up Door	West Intake	1.18 E-03	8.85 E-04	3.25 E-04	2.28 E-04	1.47 E-04
Carbon Delay Bed (East)	Diesel Building Intake	8.57 E-03	4.46 E-03	1.43 E-03	1.11 E-03	8.37 E-04
BWST (West)	Diesel Building Intake	1.86 E-02	9.65 E-03	3.08 E-03	2.39 E-03	1.84 E-03

Release Point	Receptor Point	0-2 hr X/Q	2-2.5 hr X/Q	2-8 hr X/Q	8-24 hr X/Q	1-4 days X/Q	4-30 days X/Q
Closest Main Steam Safety Valve	East Intake	1.09 E-04	9.00 E-05	4.50 E-04	1.56 E-04	9.85 E-05	8.00 E-05
Closest Atmospheric Relief Valve	East Intake	8.88 E-05	6.76 E-05	3.38 E-04	1.16 E-04	7.30 E-05	6.05 E-05
Closest Main Steam Safety Valve	CR Fire Exit Door	8.22 E-04	6.62 E-04	3.31 E-03	1.24 E-03	8.72 E-04	5.86 E-04
Closest Atmospheric Relief Valve	CR Fire Exit Door	6.98 E-04	5.58 E-04	2.79 E-03	1.02 E-03	7.54 E-04	5.45 E-04
Closest Main Steam Safety Valve	Diesel Building Intake	5.78 E-04	4.78 E-04	2.39 E-03	8.87 E-04	6.17 E-04	4.11 E-04
Closest Atmospheric Relief Valve	Diesel Building Intake	5.28 E-04	4.22 E-04	2.11 E-03	7.82 E-04	5.71 E-04	4.07 E-04