January 12, 2005

MEMORANDUM TO: James W. Clifford, Chief
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Robert L. Dennig, Chief ***/RA/***
Containment and Accident Dose Section
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

SUBJECT: SAFETY EVALUATION INPUT FOR PROPOSED LICENSE AMENDMENT FOR IMPLEMENTATION OF ALTERNATIVE SOURCE TERM AT SEABROOK STATION (TAC No. MC1097)

By letter dated October 6, 2003, FPL Energy Seabrook (FPLES), requested a license amendment for the Seabrook Station (Seabrook). The proposed amendment would revise the Seabrook licensing basis to replace the existing accident radiological source term by a full implementation of the alternative source term (AST). This change supports a future power uprate and addresses the impact of increased control room unfiltered air inleakage on control room habitability. FPLES proposes a change to the technical specification definition of dose equivalent I-131.

The Containment and Accident Dose Section of the Probabilistic Safety Assessment Branch (SPSB) has reviewed the analyses of the radiological consequences of full AST implementation. Our input to the safety evaluation for Seabrook is attached. The dose assessment of this amendment request was performed by Steve LaVie. Most of the safety evaluation (SE) was also prepared by Mr. LaVie. The SE was finalized by Jack Hayes.

Docket No. 50-443

Attachment: As stated

CONTACT: Jack Hayes, SPSB/DSSA
415-3167

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SAFETY EVALUATION INPUT BY THE PROBABILISTIC SAFETY ANALYSIS BRANCH

RELATED TO REQUEST FOR LICENSE AMENDMENT

FOR

IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

FPL ENERGY SEABROOK CORPORATION

SEABROOK STATION

DOCKETS 50-443

1.0 INTRODUCTION

By letter dated October 6, 2003, as supplemented by letters dated May 5, 2000, May 24, 2004, and July 8, 2004, FPLES requested a license amendment for Seabrook. The proposed amendment would revise the Seabrook licensing basis to replace the existing accident radiological source term by a full implementation of an AST. This change supports a future power uprate and addresses the impact of increased control room unfiltered air inleakage on control room habitability. The proposed changes are listed below:

1.1 Updated Final Safety Analysis Report

Modify the Seabrook design basis to replace the current accident source term with an AST and to replace the previous whole body and thyroid accident dose guidelines with the total effective dose equivalent (TEDE) criteria of 10 CFR 50.67(b)(2). FPLES has requested a full implementation of the AST, as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." In support of the present amendment request, FPLES did analyses of the radiological consequence of design basis accidents (DBAs) that result in offsite exposure.

Other proposed changes in the DBAs include an increase in assumed control room envelope inleakage, the development of new offsite and control room atmospheric relative concentrations (χ/Q), and an increase in the assumed rated reactor power. FPLES used dose conversion factors (DCFs) from Federal Guidance Reports (FGR) Nos. 11 and 12.

1.2 TECHNICAL SPECIFICATION Section 1.1, "Definitions"

Revise the definition of dose-equivalent I-131 in TECHNICAL SPECIFICATION Section 1.10 to replace the current reference to Regulatory Guide 1.109 with the proposed reference to FGR-11.

2.0 REGULATORY EVALUATION

In December 1999, the U.S. Nuclear Regulatory Commission (NRC) issued a new regulation, 10 CFR 50.67, "Accident Source Term," to provide a mechanism for licensed power reactors to replace the traditional accident source term used in their design basis accident analyses with an

ATTACHMENT

AST. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide 1.183. A licensee seeking to use an AST is required, pursuant to 10 CFR 50.67, to apply for a license amendment. An evaluation of the consequences of affected DBAs is required to be included with the submittal. FPLES' application of April 3, 2003, addresses these requirements in proposing to use the AST described in Regulatory Guide 1.183 as the source term in the evaluation of the radiological consequences of DBAs at Seabrook. As part of the implementation of the AST, the 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 50, Appendix A, General Design Criterion-19 (GDC-19) as the Seabrook licensing basis.

This safety evaluation 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 staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of Regulatory Guide1.183 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:

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

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

- Regulatory Guide 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 staff reviewed the technical analyses, related to the radiological consequences of design basis accidents, that were done by FPLES 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, 2004, May 24, 2004, and July 8, 2004.

The staff reviewed the assumptions, inputs, and methods used by FPLES to assess these impacts and did independent calculations to confirm the conservatism of the FPLES analyses. However, the findings of this safety evaluation input are based on the descriptions of the analyses and other supporting information submitted by FPLES. The staff also considered relevant information in the Seabrook Updated Final Safety Analysis Report (UFSAR) and the Seabrook Technical Specifications. Only docketed information was relied upon in making this safety finding.

3.1 Radiation Source Term

The Seabrook reactor core contains 193 fuel assemblies containing 492 kg 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 tonne of uranium (GWD/MTU), with a licensed peak of 62 GWD/MTU. Consistent with Regulatory Guide 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 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 FPLES 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 technical specifications. FPLES 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 technical specification limit of 1.0 μci/gm dose equivalent I-131. The proposed definition of dose equivalent I-131 and the thyroid dose conversion factors of FGR-11 were used in this adjustment. Non-iodine species were normalized to the technical specification limit of $100/\bar{E}\gamma$.

The secondary coolant specific activity technical specification 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 FPLES approach to be consistent with regulatory guidance and staff practice.

The intent of the technical specifications 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 dose conversion factors, 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 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. FPLES 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. FPLES utilized this distribution and the inhalation thyroid dose conversion factors from FGR-11 to calculate the activity level of isotopes ^{131}I - ^{135}I at an overall primary coolant activity level of 1μCi/g of dose equivalent 131I. The licensee utilized this activity level to calculate the dose consequences of main steamline break (MSLB) and steam generator tube rupture (SGTR) accidents at 1 μCi/g and at 60 μCi/g dose equivalent ^{131}I. TEDE doses were calculated using effective dose equivalent (EDE) dose conversion factors from FGR-11. The results met the acceptance criteria of Regulatory Guide 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 could accept FPLES' approach using the thyroid dose conversion factors but the licensee's proposed definition of dose equivalent ^{131}I needed to be changed to reflect the actual manner of calculation.

FPLES' proposed definition included the sentence, "DOSE EQUIVALENT 1-131 shall be that concentration of 1-131 (micro curie per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of 1-131, 1-132, 1-133, 1-134, and 1-135 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 FPLES definition did not specify what thyroid dose conversion factors from FGR-11 should be used. As proposed, it could have been the thyroid dose conversion factors from any of the Tables for inhalation, ingestion or submersion. Since the thyroid dose conversion factors 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 ___, 2004, request for additional information (RAI) to FPLES, the following definition of dose equivalent I-131 was proposed as reflecting the methodology actually used:

> DOSE EQUIVALENT 1-131 shall be that concentration of 1-131 (micro curie per gram) which alone would produce the same thyroid TEDE dose as the quantity and isotopic mixture of 1-131, 1-132, 1-133, 1-134, and 1-135 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 FPLES' January ___, 2005, response to the RAI, they indicated their acceptance of this proposed definition and proposed the definition as noted above.

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 (χ/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 Regulatory Guide 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) χ/Q values for the loss-of-coolant accident (LOCA), fuel handling accident (FHA), main steamline break (MSLB), steam generator tube rupture (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. Stable and neutral conditions were consistently reported to occur at night and unstable and neutral conditions during the day, as expected. 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 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 making estimates of atmospheric dispersion estimates for design basis accident assessments.

3.3.2 CR Atmospheric Dispersion Factors

The licensee made numerous CR χ/Q calculations using guidance from Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," and provided χ/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 χ/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 control room χ/Q values are presented in Table 2. 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 χ/Q values calculated using ARCON96 by a factor of five, consistent with guidance in Regulatory Guide 1.194 for postulated releases from steam relief and atmospheric dump valves that meet the specified criteria. Among the criteria, Regulatory Guide 1.194 states that the time-dependent vertical velocity must exceed the 95th-percentile wind speed at the release point height by a factor of at least five. The licensee estimated the 95th-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. Staff made confirmatory estimates from the 1998 through 2002 meteorological data and concluded that the licensee 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 95th-percentile wind speeds is greater than a factor of five.

- As addressed in Regulatory Guide 1.194, the licensee also reduced χ/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 control room air intakes, the χ/Q values for the limiting intake were reduced by a factor of two since flow into the intakes is equal. The χ/Q values for the postulated RWST release to the diesel intakes are averages based upon Equation 5a of Regulatory Guide 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 χ/Q values in Table 2 are acceptable for use in design basis accident CR dose assessments.

3.3.3 EAB/LPZ Atmospheric Dispersion Factors

The licensee calculated EAB and LPZ χ/Q values using the 1998 through 2002 onsite meteorological data and the PAVAN computer code which implements the guidance provided in Regulatory Guide 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. Staff qualitatively reviewed the inputs to the PAVAN computer runs and found the inputs generally consistent with site configuration drawings and staff practice. However, staff notes that the EAB χ/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 χ/Q value only.

In summary, the 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 χ/Q values (except for the EAB χ/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 χ/Q values are acceptable for use in design basis accident EAB and LPZ dose assessments.

3.4 Accident Dose Calculations

In accordance with the guidance in Regulatory Guide 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, FPLES did an evaluation of previously analyzed DBAs to decide which, if any, were affected by the proposed amendment. FPLES re-analyzed the radiological consequences of the following DBA events:

- Loss-of-coolant accident (LOCA)
- Fuel handling accident (FHA)
- Main steamline break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Reactor Coolant Pump Shaft Seizure (LRA)
- Rod Cluster Control Assembly (RCCA) Ejection
- Letdown Line Break (LLB)
- Radioactive Gaseous Waste System Leak or Failure (GWL)
- Radioactive Liquid Waste System Leak or Failure (LWL)

3.4.1 Loss of Coolant Accident (LOCA)

The accident considered is 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 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 Regulatory Guide 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

FPLES 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. FPLES 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 FPLES 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, FPLES projects that 85 percent of the containment free volume is sprayed. FPLES 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. FPLES 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). FPLES 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, which time the removal rate is decreased to 0.575 hr^{-1}.

FPLES assumes that the radioiodine released to the containment atmosphere consists of 95 percent 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 Regulatory Guide 1.183 and are, therefore, acceptable.

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 refueling water storage tank (RWST) due to ECCS backleakage.

Containment Purge at Event Initiation

FPLES 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, FPLES 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 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. A containment enclosure, consisting of a reinforced concrete cylindrical structure with a hemispherical dome, surrounds the containment. Fans maintain the pressure in the space between the containment structure and the containment enclosure 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. FPLES 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. FPLES assumes that after 8 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 Technical Specification 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. FPLES 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. At about 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. FPLES assumes that the leakage rate is two times the expected value, or 48 gallons per day.

FPLES 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. FPLES deviated from the guidance of Regulatory Guide 1.183 and proposed the following alternative treatment of the radioiodine release from the leaked ECCS liquid:

- FPLES 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.

- FPLES 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.

Regulatory Guide 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. FPLES assumptions result in only 1.15 percent of the total iodine flashing. The staff challenged the FPLES assumptions in an RAI. FPLES responded informally on April 19, 2004. In a teleconference on April 20, 2004, the staff stated their objection to the proposed response. FPLES submitted a formal response on May 24, 2004. In this response, FPLES provided a justification for their proposed alternative to the Regulatory Guide 1.183 guidance, and provided the results of an analysis done using the Regulatory Guide 1.183 assumption. The FLPES argument was not persuasive and the staff has based this safety evaluation on the revised analysis that used the Regulatory Guide 1.183 assumptions regarding the release of radioiodine. Although FPLES 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 a January ___, 2005, response to a December ___, 2004, RAI, FPLES 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.

Release Due to ECCS Back Leakage to the RWST

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. FPLES 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. FPLES assumes that the radioiodine speciation in the RWST would be 99 percent aerosol and 1.0 percent elemental. FPLES 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 evaporation to dryness is not likely. This elemental radioiodine is assumed to become volatile and will partition between the liquid and vapor space in the RWST. FPLES 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. FPLES 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) FPLES 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) FPLES 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 Control Room Modeling

The control room normal ventilation isolation will be activated by a high containment pressure signal at event initiation. FPLES 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. FPLES 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 test, which showed Train A inleakage to be 8±11 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 FPLES Generic Letter 2003-01 response. That response is being evaluated as a separate proceeding.

Summary–LOCA

The staff found that FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for the LOCA were found to meet the applicable accident dose criteria and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

3.4.2 Fuel Handling Accident (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 depending on their physical and chemical form. Appendix B of Regulatory Guide 1.183 identifies acceptable radiological analysis assumptions for a FHA.

FPLES assumed no decontamination for noble gases, an effective decontamination factor of 200 for radioiodines, and retention of all aerosol and particulate radionuclides. FPLES assumed that 100 percent of the radionuclides released from the reactor cavity or spent fuel pool are released to the environment in two hours without any credit for filtration, holdup, or dilution. 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, FPLES did the analysis using the atmospheric dispersion factors for most limiting combination of release point and receptor.

FPLES 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 that are greater than two times background on GM radiation detectors located in the ventilation intake ductwork. FPLES 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. FPLES 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 FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for the FHA were found to meet the applicable accident dose criteria and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

3.4.3 Main Steamline Break (MSLB)

The accident considered is the complete severance of a main steam line 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 Regulatory Guide 1.183 identifies acceptable radiological analysis assumptions for an MSLB.

FPLES 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 technical specifications. 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 8 hours.

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

Leakage from the RCS to the steam generators is assumed to be the maximum value permitted by technical specifications. 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. FPLES 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 degrees (at about 48 hours). The Seabrook surveillance procedures used to demonstrate compliance with RCS technical specification leakage normalize the temperature of all leakage streams to temperatures consistent with normal power operation conditions. In converting the technical specification maximum allowable volumetric flow to mass flow values for input to analyses, FPLES uses a density value consistent with normal power operation conditions. This is consistent with the guidance of Regulatory Guide 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. FPLES determined that the tubes in the unaffected steam generators would remain covered by the bulk water. FPLES 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.

FPLES 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 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. FPLES 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 staff found that FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for the MSLB were found to meet the applicable accident dose criteria and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

3.4.4 Steam Generator Tube Rupture (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 Regulatory Guide 1.183 identifies acceptable radiological analysis assumptions for an SGTR.

FPLES 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 technical specifications. 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 8 hours.

For the two analyzed cases, FPLES 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. FPLES 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, FPLES assumes there is primary -to-secondary leakage at the maximum value permitted by technical specifications. 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. FPLES 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 degrees (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. FPLES determined that tubes in the unaffected steam generators would remain covered by the bulk water. FPLES assumes that the radionuclide concentration in the steam generator is partitioned such that 1 percent of the radionuclides in the steam generators 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.

FPLES 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 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. FPLES 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 staff found that FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for the SGTR were found to meet the applicable accident dose criteria and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

3.4.5 Reactor Coolant Pump Shaft Seizure (LRA)

The accident considered is the instantaneous seizure of a reactor coolant pump rotor (i.e., a locked rotor accident) which 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 Regulatory Guide 1.183 identifies acceptable radiological analysis assumptions for an LRA.

FPLES 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 technical specification value of 500 gpd for any steam generator and 1.0 gpm for all steam generators for eight hours. FPLES 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. FPLES assumes that the radionuclide concentration in the steam generator is partitioned such that 1 percent of the radionuclides in the bulk water of the steam generators 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.

FPLES 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 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. FPLES 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 staff found that FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for the LRA were found to meet the applicable accident dose criteria and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

3.4.6 Rod Cluster Control Assembly (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 were the only pathway, would show that the composite dose would also be acceptable. Appendix H of Regulatory Guide 1.183 identifies acceptable radiological analysis assumptions for an RCCA ejection.

FPLES 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. FPLES 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. FPLES assumes after 8 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. FPLES does not assuming mixing of the containment leakage in the containment enclosure. FPLES does not credit containment spray operation as a radionuclide removal mechanism. However, FPLES 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 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 FPLES 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.

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. FPLES 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. FPLES conservatively assumed that the chemical form of the radioiodine released to the environment would be 97 percent elemental and 3 percent organic. The FPLES 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.

FPLES 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 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, FPLES 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, FPLES 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 FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for both cases of the RCCA ejection were found to meet the applicable accident dose criteria and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

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. FPLES also assumes a release from the secondary plant via the main condenser. This event is

not specifically addressed in Regulatory Guide 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 technical specifications. 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 8 hours.

FPLES 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, FPLES 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, FPLES stated that they were conservatively assuming that radioiodine partitioning was not applicable to steam generators at power. Instead, FPLES assumed a decontamination factor of 100 for radioiodines and aerosols in the main condenser. FPLES 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 staff notes that the Seabrook UFSAR analysis for the LLB does not address releases via the secondary system. Regulatory Position 5.1.2 of Regulatory Guide 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 staff issued RAI No. 6 in their December ___, 2004, letter which addressed the release via the condenser for the letdown line rupture. In response to the release via the condenser, FPLES' January ___, 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 FPLES submittal. It was misstated.

In addition to RAI No. 6, the staff also requested additional information concerning the FPLES' assumption of no reactor trip and no loss of offsite power for this event. This was contained in RAI Nos. 4 and 5 in the December ___, 2004, letter. In response to those requests, FPLES 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 RAI 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 FPLES has maximized the releases by assuming no loss of offsite power and no reactor trip, the incorporation of these assumptions into the letdown line rupture event is acceptable.

FPLES 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 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. FPLES 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 FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for the LLB were found to meet the applicable accident dose criteria and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

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 Regulatory Guide 1.183, in an SRP chapter or a regulatory guide. 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.

FPLES assumes that the delay beds contain the design inventories of radioactive gases that are based on long-term plant operation with 1 percent failed fuel. FPLES assumes that the entire inventory is released at an arbitrary, but conservative, rate of 10,000 ft^3 in two hours.

FPLES 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. FPLES 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 Regulatory Guide 1.183 does not address this event, FPLES 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. FPLES 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 BTP 11-5, the acceptance criterion used for consequences of the release of the contents of the offgas system from a 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 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, which exceeds a small fraction of Part 100.

In the December ___, 2004, request for additional information the 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, FPLES provided the following in their January ___, 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 FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for the GWL were found to meet the applicable accident dose criteria, as discussed above, and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

3.4.9 Radioactive Liquid Waste System Leak or Failure (LWL)

The accident considered is the release of fission gases from an unexpected and uncontrolled release of 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 Regulatory Guide 1.183, in an SRP chapter or in a regulatory guide. The staff relied upon Seabrook's current licensing basis for this review.

FPLES 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. FPLES assumes that the entire gaseous inventory is released at an arbitrary, but conservative, rate of 10,000 ft^3 in two hours.

FPLES 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. FPLES 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 Regulatory Guide 1.183 does not address this event, FPLES 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 December ___, 2004, the staff issued a request for additional information (No. 8) concerning the criterion for this event. In response to that request, FPLES 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 10CFR 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."

The staff finds this change in acceptance criterion for this event acceptable.

The staff found that FPLES 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. The EAB, LPZ, and control room doses estimated by FPLES for the LWL were found to meet the applicable accident dose criteria, as discussed above, and are, therefore, acceptable. The staff did independent calculations and confirmed the FPLES conclusions.

3.5 Technical Specification 1.1, "Definitions," Change

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

The intent of the technical specifications 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 dose conversion factors, 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 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. FPLES 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. FPLES utilized this distribution and the inhalation thyroid dose conversion factors 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 of dose equivalent 131I. 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 dose conversion factors from FGR-11. The results met the acceptance criteria of Regulatory Guide 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 could accept FPLES' approach using the thyroid dose conversion factors but the licensee's proposed definition of dose equivalent ^{131}I needed to be changed to reflect the actual manner of calculation.

FPLES' proposed definition included the sentence, "DOSE EQUIVALENT 1-131 shall be that concentration of 1-131 (micro curie per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of 1-131, 1-132, 1-133, 1-134, and 1-135 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 FPLES definition did not specify what thyroid dose conversion factors from Federal Guidance Report 11 should be used. As proposed, it could have been the thyroid dose conversion factors from any of the Tables for inhalation, ingestion or submersion. Since the thyroid dose conversion factors 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 ___, 2004, RAI to FPLES, the following definition of dose equivalent I-131 was proposed as reflecting the methodology actually used:

> DOSE EQUIVALENT 1-131 shall be that concentration of 1-131 (micro curie per gram) which alone would produce the same thyroid TEDE dose as the quantity and isotopic mixture of 1-131, 1-132, 1-133, 1-134, and 1-135 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 FPLES' January ___, 2005, response to the RAI, they indicated their acceptance of this proposed definition and proposed the definition as noted above. The staff finds FPLES proposed definition of January ___, 2005, acceptable.

4.0 STATE CONSULTATION

SPSB has no input.

5.0 ENVIRONMENTAL CONSIDERATION

SPSB has no input.

6.0 CONCLUSION

As described above, the staff reviewed the assumptions, inputs, and methods used by FPLES to assess the radiological impacts of the proposed full implementation of an AST at the Seabrook Station. The staff finds that FPLES 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 FPLES 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 control room doses will comply with these criteria. The staff finds reasonable assurance that the Seabrook Station, as modified by this proposal, 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 design basis accidents.

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 FPLES. The previous offsite and control room 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 Regulatory Guide 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 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.

7.0 REFERENCES

As stated in text

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Date: January 5, 2005

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

<u>ECCS Leakage Pathway</u>

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 Fuel Handling Accident 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 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 steam line 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

Assumptions for Control Rod Ejection Accident Analyses

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 & 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

Letdown Line Rupture

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

Waste System Failure

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