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June 25, 2009

L-2009-133
10 CFR 50.90

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

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
License Amendment Request 196
Alternative Source Term and Conforming Amendment

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) hereby requests an amendment to Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4. FPL proposes to revise the Turkey Point Units 3 and 4 licensing bases to adopt the alternative source term (AST) as allowed in 10 CFR 50.67

Enclosure 1 provides a description of the proposed change and supporting justification including the determination of no significant hazards and environmental considerations. The Attachments to the Enclosure are as follows: Attachment 1 provides the existing TS pages marked up to show the proposed changes. Attachment 2 provides existing TS Bases pages marked up to show the proposed changes. Attachment 3 provides Numerical Applications, Inc., NAI-1396-045, "AST Licensing Technical Report for Turkey Point Units 3 and 4," Revision 1. Attachment 4 provides a summary of the regulatory commitments made in this submittal. Attachment 5 contains a comparison of the inputs and assumptions from the Turkey Point current licensing bases with the inputs and assumptions used for the AST. Attachment 6 contains a Regulatory Information Summary (RIS) 2006-04 compliance matrix. Attachment 7 contains a list of plant modifications required to support the AST.

FPL requests approval and issuance of the proposed amendments by June 30, 2010.

The Turkey Point Plant Nuclear Safety Committee has reviewed the proposed license amendments. In accordance with 10 CFR 50.91(b)(1), a copy of the proposed amendments is being forwarded to the State Designee for the State of Florida.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c). FPL has determined that the proposed changes do not involve a significant hazards consideration.

A001
NRR

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on June 25, 2009.

Very truly yours,

William Jefferson, Jr.
Site Vice President
Turkey Point Nuclear Plant

Attachments

cc: Regional Administrator, Region II, USNRC
USNRC Project Manager, Turkey Point Nuclear Plant
Senior Resident Inspector, USNRC, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health

TABLE OF CONTENTS

LICENSE AMENDMENT REQUEST
LAR 196

ALTERNATIVE SOURCE TERM AND CONFORMING AMENDMENT

ENCLOSURE 1

TABLE OF CONTENTS

LICENSE AMENDMENT REQUEST

ALTERNATIVE SOURCE TERM AND CONFORMING AMENDMENT

1.0 Purpose and Scope

Florida Power and Light Company (FPL) proposes to revise the Turkey Point (PTN) Units 3 and 4 licensing bases to adopt Alternative Source Term (AST) as allowed pursuant to 10 CFR 50.67 (Reference 1). Sections 3.0 and 4.2 and Attachment 1 provide the descriptions and justifications for the Technical Specifications (TS) being revised to support the AST analyses. Attachment 2 provides the associated TS Bases, supplied for information only, to support the NRC review. Implementation of AST will allow an increase in the design basis assumption regarding control room envelope unfiltered inleakage, while maintaining the accident dose consequences within the limits specified by 10 CFR 50.67.

The input values assumed in the AST analyses bound both the current plant operation and a planned extended power uprate (EPU). This LAR is requesting approval of AST for dose consequences and is not asking for approval of a licensed thermal power uprate.

Additionally, a new TS addressing the use of sodium tetraborate decahydrate (NaTB) for containment sump pH control will be added. This TS and its associated Limiting Condition for Operation (LCO), Actions, and Surveillance Requirements provide assurance that the proper amount of NaTB is maintained in the containment to ensure that the containment sump pH is at the value assumed by the AST loss of coolant accident (LOCA) radiological consequence analysis.

2.0 Background Information

The PTN Units 3 and 4 current licensing bases for the radiological consequences analyses of accidents other than the fuel handling accident (FHA) are based on source methodologies and assumptions derived from Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (Reference 3). The accidents are discussed in the Chapter 14 of the Turkey Point Updated Final Safety Analysis Report (UFSAR) (Reference 2). The current PTN dose consequences of design basis events are based on the criteria stated in 10 CFR Part 100 and 10 CFR 50, Appendix A General Design Criterion (GDC) 19. The current licensing basis for the FHA radiological consequence analysis is 10 CFR 50.67, Accident Source Term, is being re-analyzed to address the planned EPU conditions.

PTN predates the NRC issuance of the 10 CFR 50 Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants." The PTN GDCs identified in the UFSAR were based on the Atomic Energy Commission (AEC) 1967 proposed GDCs. However, PTN has committed to 10 CFR 50, Appendix A, GDC 19 - Control Room. The AST analyses meet the criteria stated in 10 CFR 50.67 and 10 CFR 50 Appendix A, GDC 19 - Control Room.

10 CFR 50.67 was issued by the NRC to permit holders of facility operating licenses to voluntarily revise the traditional accident source term used in the design basis accident radiological consequence analyses with AST. 10 CFR 50.67 requires a licensee seeking to use the AST to apply for a license amendment and requires that the application include an evaluation of the consequences of affected design basis accidents. Regulatory guidance for the implementation of the AST is provided in NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" and NRC Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Terms," (References 4 and 5, respectively).

NEI-99-03, "Control Room Habitability Guidance" (Reference 6) indicated that several nuclear power plants demonstrated control room unfiltered inleakage rates in excess of amounts assumed by the radiological consequence analyses during testing. To address this issue, NRC Generic Letter (GL) 2003-01, "Control Room Habitability" (Reference 7) required utilities to assess the most limiting unfiltered inleakage into the control room envelope. The PTN response to GL 2003-01 (Reference 8) confirmed that the PTN control room envelope and ventilation system meet all applicable habitability regulations and are designed, constructed, configured, operated, and maintained in accordance with the current licensing basis, as described in the PTN UFSAR.

The full implementation of the AST methodology is being used to calculate the offsite and control room radiological consequences for PTN Units 3 and 4 for events discussed in RG 1.183. Attachment 3, "NAI-1396-045 – AST Licensing Technical Report for Turkey Point Units 3 and 4" (Reference 9). Section 2.0 describes the compliance with RG 1.183 Regulatory Positions for each of the accidents listed below.

The following limiting UFSAR Chapter 14 accidents are analyzed:

- Loss-of-Coolant Accident (LOCA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Locked Rotor
- Rod Cluster Control Assembly (RCCA) Ejection
- Fuel Handling Accident (FHA)
- Spent Fuel Cask Drop, and
- Waste Gas Decay Tank (WGDT) Rupture

RG 1.183 does not address the Spent Fuel Cask Drop or the WGDT Rupture Events. For the Spent Fuel Cask Drop event, the FHA dose limits in RG 1.183 have been adopted. For the WGDT Rupture event, the Branch Technical Position 11-5 of the Standard Review Plan (SRP) gives dose acceptance criteria for the exclusion area boundary (EAB) to 100 mrem. RIS 2006-04 establishes the AST dose limit to a member of the public for this event at 100 mrem TEDE. Since RIS 2006-04 associates this acceptance criteria with the annual limit established in 10 CFR 20, the EAB dose will be evaluated over the duration of the event rather than for the worst two-hour period. Branch Technical Position 11-5 does not require the dose consequences to be evaluated at the LPZ or the control room. Although no limit is given for the low population zone (LPZ), the LPZ is evaluated against this 0.1 rem TEDE limit. A TEDE limit of 5.0 rem stated in 10 CFR 50.67 is applied to the control room consistent with other AST events. All doses are evaluated for a 30-day period.

RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss–of-Coolant Accident," indicates that the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodine, if the containment sump pH is controlled at values of 7.0 or greater, consistent with the recommendation in RG 1.183.

PTN will install a Recirculation Sump pH Control System that will ensure that the pH is controlled at 7.0 or greater prior to the onset of containment spray recirculation. The system will consist of ten stainless steel wire mesh baskets containing NaTB in the lower regions of containment and is described in Section 4.2.10. The system will ensure that the pH is maintained at 7.0 or greater for the entire recirculation spray cycle.

3.0 Description of Proposed Changes

This license amendment request (LAR) revises the PTN Units 3 and 4 licensing basis to fully implement AST, as described in RG 1.183, Section 1.2, through reanalysis of the radiological consequences of the UFSAR Chapters 11 and 14 accidents listed above.

Full implementation of AST for PTN Units 3 and 4 does not include Environmental Qualification (EQ) of safety related equipment inside the containment, as described in Section 4.2 in this LAR. RG 1.183, Positions 1.3 and 4.3 identify various licensing requirements for which compliance may have been demonstrated, in part, by the evaluation of radiological consequences of design basis accidents. Since this amendment is not requesting approval of a thermal power increase, the current source term and the PTN responses to these licensing commitments are not being revised as a result of implementation of AST remain applicable. In addition, Vital Area Access dose and shielding calculations are unchanged with the implementation of AST.

The following changes to the licensing basis are assumed in the supporting analyses:

- 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, GDC 19.
- New onsite (Control Room) and offsite (EAB and LPZ) atmospheric dispersion factors (X/Q) have been determined using current meteorological data following the guidance in RG 1.194 and 1.145 (Reference 12 and 13, respectively).
- Dose conversion factors for inhalation and submersion use Environmental Protection Agency (EPA) Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion

Factors for Inhalation, Submersion, and Ingestion," and EPA Federal Guidance Report No. 12 (FGR 12) "External Exposure to Radionuclides in Air, Water, and Soil" (References 10 and 11, respectively).

- Control room unfiltered inleakage limits were increased until the dose limit was approached for LOCA. This unfiltered leakage value is assumed for all other events.
- A more restrictive TS primary coolant specific activity limit for DOSE EQUIVALENT I-131 (DEI-131) is utilized.
- The primary-to-secondary SG leakage rate assumed in the accident analysis and specified in the accident induced leakage performance criterion of TS 6.8.4.j.b.2, "Steam Generator Program," is reduced from "1 gpm through all steam generators and 500 gpd through any one steam generator" [at accident conditions] to "0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions." The current primary to secondary operational leakage limit remains unchanged.
- A more restrictive TS containment leakage is utilized.
- Containment sump pH control is provided by NaTB stored in ten stainless steel baskets inside containment.
- The Containment Emergency Filtration System is no longer credited to maintain the AST dose consequence analyses within regulatory limits.
- The Containment Spray System and the proposed Recirculation pH Sump Control System are being credited for post-LOCA iodine removal.
- For the Secondary RCCA Ejection, FHA within the Fuel Building, and Spent Fuel Cask Drop events, credit is taken for the operators to manually initiate the emergency recirculation mode of the Control Room Ventilation System within 30 minutes.
- The Control Room Emergency Ventilation System intakes are being relocated to diverse wind sectors for post-accident contaminants.
- The methyl iodide penetration criterion is more restrictive than the current TS limit for the control room emergency ventilation filter.

Accordingly, the following changes are proposed to the PTN TS. Justification for these proposed changes is provided in Section 4.2: (Attachment 1 provides the marked-up TS pages for the changes listed below.)

- <u>TS 1.12</u>, Definition of DOSE EQUIVALENT I-131 is revised to reference EPA FGR 11, as the source of thyroid dose conversion factors.
- <u>TS 1.13</u>, Definition of E-AVERAGE DISINTEGRATION ENERGY is deleted and replaced with the definition of DOSE EQUIVALENT XE-133 that includes a reference to EPA FGR 12 as the source of whole body dose conversion factors.

- TS 3.4.8, Reactor Coolant System Specific Activity
  - LCO 3.4.8 a., RCS specific activity limit for DOSE EQUIVALENT I-131 is reduced from "less than or equal to 1.0 microcurie per gram" to "less than or equal to 0.25 microcuries per gram."
  - LCO 3.4.8 b., RCS specific activity limit is changed from "less than or equal to 100/Ē microcuries per gram of gross radioactivity" to "less than or equal to 447.7 microcuries per gram DOSE EQUIVALENT XE-133."
  - APPLICABILITY is changed from "MODES 1, 2, 3, 4, and 5" to "MODES 1, 2, 3, and 4."
  - ACTION MODES 1, 2, and 3*, a. and b. and ACTION MODES 1, 2, 3, 4, and 5, are deleted and replaced with new ACTIONS a., b., c., d., and e.

    a. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is less than or equal to 60 microcuries per gram once per 4 hours.
    b. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 60 microcuries per gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 0.25 microcuries per gram limit. Specification 3.0.4 is not applicable.
    c. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131 for greater than or equal to 48 hours during one continuous time interval, or greater than 60 microcuries per gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.
    d. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 447.7 microcuries per gram limit. Specification 3.0.4 is not applicable.
    e. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133 for greater than or equal to 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.
  - Footnote on page 3/4 4-26 is deleted.

- FIGURE 3.4-1, DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1μCi/gram DOSE EQUIVALENT I-131 - This figure is deleted and replaced with "This page is intentionally left blank."
- TABLE 4.4-4, Item 1 is deleted and replaced with "NOT USED."
- TABLE 4.4-4, Item 3 is changed
  - from "Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration" to "Isotopic Analysis for DOSE EQUIVALENT I-131,"
  - from "1 per 14 days" to "a) 1 per 14 days" and "b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period."
- TABLE 4.4-4, Item 5 is changed
  - from "Radiochemical for Ē Determination" to "Isotopic Analysis for DOSE EQUIVALENT XE-133,"
  - from "1 per 6 months*" to "1 per 7 days,"
- TABLE 4,4-4, Item 6, Isotopic Analysis for Iodine Including I-131, I-133, and I-135 is deleted and replaced with "NOT USED."
- TABLE NOTATIONS are deleted and replaced with "THIS PAGE INTENTIONALLY BLANK."

- TS 3/4.6.3, Emergency Containment Filtering System is deleted, and replaced with "Not Used."
- TS 3/4.7.5, Control Room Emergency Ventilation System Surveillance Requirement 4.7.5.c 2, the methyl iodide penetration criteria is reduced from "less than 2.5%" to "less than 1.25%."
- TS 3.7.9, Gas Decay Tanks LCO is clarified to change the phrase within the parentheses from "(considered as Xe-133 equivalent)" to "(DOSE EQUIVALENT XE-133)".
- TS 6.8.4.h, Containment Leakage Rate Testing Program is changed to reduce the maximum allowable containment leakage rate acceptance criterion from "0.25% of containment air weight per day" to "0.20% of containment air weight per day."
- TS 6.8.4.j.b.2, SG Program is revised to change the accident-induced performance criterion from "1 gpm total through all steam generators and 500 gallons per day through any one steam generator" [at accident conditions] to "0.60 gpm total through all steam generators and 0.20 gpm through any one steam generator at room temperature conditions."

- TS 6.9.1.2, Annual reports is changed to address the change in RCS specific activity discussed above. This change addresses the change in the LCO associated with the DOSE EQUIVALENT I-131 being reduced from "1.0 microcurie per gram" to "0.25 microcuries per gram." The footnote "** This tabulation supplements the requirements of §20.2206 of 10 CFR Part 20" at the bottom of the page is deleted. There is no text on the page that requires this footnote.

The following new TS is created:

- TS 3/4.6.2.3, Recirculation pH Control System, This is a new TS to control the post-LOCA containment sump solution pH using NaTB. This passive system will consist of ten stainless steel baskets containing NaTB in the lower regions of the containment.

4.0 Basis/Justification for the Proposed Changes

4.1 AST Compliance to Regulatory Guide 1.183

Attachment 3 Section 2.0 describes the compliance with RG 1.183 Regulatory Positions for each of the accidents being analyzed.

4.1.1 Accident Source Term

The full core isotopic inventory for PTN Units 3 and 4 is determined in accordance with RG 1.183. The bounding values for the core thermal power and fuel parameters were based on a planned extended power uprate (EPU) value. The fission product inventory in the core and coolant systems available for release to the containment is based on the maximum full power operation of the core and the proposed licensed values for fuel enrichment and fuel burnup that equal or bound current licensed values. This isotopic source term bounds the licensed thermal power. This LAR is only requesting approval of AST for dose consequences at a power level above the current licensed power limit. This amendment request does not seek approval to operate at a power level beyond the current licensed power.

NAI-1396-045, "AST Licensing Technical Report for Turkey Point Units 3 and 4," (Attachment 3) provides the details of the LOCA and non-LOCA accident dose consequence analyses performed according to the guidelines set forth in RG 1.183. The radiation source terms for PTN are calculated as described in Attachment 3, Section 1.7.

The fission product inventory and assembly source term are based on a bounding maximum core power of 2644 MWt with a calorimetric uncertainty of 0.3% or 2652 MWt which exceeds the current licensed rated thermal power of 2300 MWt. For non-LOCA events with fuel failures, a radial peaking factor of 1.65 is applied to conservatively simulate the effect of power level differences across the core that might affect the localized fuel failures for assemblies containing the peak fission product inventory.

The primary source term methodology described in Attachment 3, Section 1.7.2 is based on maximum equilibrium concentrations from operation at 2652 MWt with defects in one percent of the fuel rod cladding. The equilibrium iodine activities were adjusted to achieve the proposed TS 3.4.8 DEI-131 limit of 0.25 microcuries per gram using the proposed TS definition for DEI-131.

Secondary system coolant activity is limited to a value of less than or equal to 0.10 microcuries per gram DEI-131 in accordance with TS 3.7.1.4. Noble gases entering the secondary system coolant are assumed to be immediately released; thus the noble gas activity concentration in the secondary system coolant is assumed to be 0.0 microcuries per gram. As such, the secondary side iodine activity is 40% of the primary coolant activity (the ratio of 0.1 microcuries per gram to 0.25 microcuries per gram, i.e., the ratio of the two TS limits of the secondary side coolant source term to the primary side coolant source term.)

4.1.2 Dose Calculation

The PTN radiological dose calculations were performed using the AST methodology applying TEDE acceptance criteria. Dose calculations follow the guidelines of Regulatory Positions cited in RG 1.183. Conformance with these regulatory positions is described in Attachment 3, Section 2.0 for each of the accidents analyzed. Offsite and control room doses meet the guidelines of RG 1.183 and the requirements of 10 CFR 50.67. RG 1.183 does not address the Spent Fuel Cask Drop and WGDT rupture events. The criteria adopted for these events are described in Section 4.1.5 below.

Analyses consider the radionuclides listed in Table 5 of RG 1.183. The chemical form of the radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine and 0.15% organic iodine, including gap releases and fuel pellet releases. With the exception of elemental, organic iodine, and noble gases, fission products are assumed to be in particulate form. The accident specific transport of these iodine species are described in Attachment 3 Section 2.0.

Increased values for control room unfiltered inleakage where determined by increasing the unfiltered inleakage assumption in the LOCA dose calculation until the dose limit was approached. This unfiltered leakage value was assumed for all other events.

4.1.3 Atmospheric Dispersion (X/Q) Factors

X/Q factors for various release-receptor combinations were developed to replace previously approved values used for control room and offsite dose consequences. The X/Q factors were determined in accordance with the guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," and RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (References 12 and 13, respectively).

The Control Room Emergency Ventilation outside air intakes are being relocated from their current locations to new locations that fall within diverse wind sectors for post-accident contaminants. See Attachment 3, Figure 1.8.1-1. The relocated intakes will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR 50 Appendix A, GDC 19 to satisfy the operability requirements for the control room emergency ventilation system. In the emergency/ recirculation mode of operation, the intakes will have balanced intake flow rates and capable of drawing outside makeup air from both analyzed intake locations. The intakes will be located near the ground level beyond the southeast and northeast corners of the auxiliary building. In accordance with RG 1.194 (Reference 12), the dispersion factors for the most conservative intake are selected and the dual intake dilution credit allows the X/Qs to the most limiting emergency intake to be reduced by a factor of two in the event analyses for all release points in separate wind sectors. Since the control room envelope remains pressurized while in the emergency/recirculation mode, unfiltered inleakage is modeled as makeup air which bypasses the filtration system. Modeling of the control room conservatively addresses these factors, as they apply to the various release locations for each analyzed event.

A number of release-receptor combinations are considered for the onsite control room atmospheric dispersion factors. The different cases are used to determine the limiting release-receptor combination for the analyzed events. New X/Q factors are also developed for offsite receptor locations. The EAB offsite dose evaluation conservatively applies the worst case two-hour X/Q for the entire event to the worst two-hour dose contribution from different time periods. Attachment 3, Section 1.8 identifies the codes used and details the methodology and assumptions used in developing the new X/Q factors.

Meteorological data over a five-year period (2003 through 2007) is used in the development of the new atmospheric dispersion (X/Q) factors used in the analysis. This is described in detail in Section 1.8.3 of Attachment 3. The data is being provided in a separate submittal to facilitate NRC review of the models used in the analysis.

4.1.4 Assumptions and Methodologies

The AST analyses performed for PTN Units 3 and 4 use assumptions and models defined in RG 1.183 to provide appropriate and prudent safety margins. Attachment 5 provides a comparison of the current dose consequence analyses inputs/assumptions to the new AST dose consequence analyses for each of the events being analyzed. The current PTN dose consequences analyses, excluding FHA, use the TITAN5 computer code. The current FHA dose consequences analysis uses the RADTRAD-NAI computer code.

Section 2.0 of Attachment 3 describes the specific assumptions for each radiological consequence analysis, the methodology, transport inputs (e.g. ventilation), removal inputs (e.g. reduction in airborne radioactivity due to containment spray), and the radiological consequences. Selected numeric input values use the limiting or bounding input for either the current power or the uprated power to assure a conservative calculated dose.

The following computer codes are used in performing the AST analyses:

- ARCON96
- Microshield
- ORIGEN
- PAVAN
- GOTHIC
- RADTRAD-NAI

Attachment 3, NAI-1396-045, Section 1.5 provides a discussion of the function of each computer code. Each of the codes has been used in previously approved implementations of AST, including the St. Lucie Unit 1 LAR and Amendment (References 14 and 15, respectively).

Overfill is not assumed to occur as a result of a steam generator tube rupture event. This design basis assumption has been previously confirmed for Turkey Point for the current licensed power conditions by a supplemental analysis using the LOFTTR2 computer code demonstrating margin to overfill following a postulated steam generator tube rupture event assuming extended operator action times for break flow termination. A similar confirmatory analysis demonstrated margin to overfill for the planned EPU conditions. These analyses used a computer code (LOFTTR2) and followed a methodology previously approved by the NRC for the SGTR event utilizing modeling assumptions that are consistent with the Turkey Point Licensing basis (Reference 35).

4.1.5 Dose Consequences Results

Analysis of the dose consequences for the LOCA, FHA, MSLB, SGTR, Locked Rotor, and RCCA Ejection Events are made using the RG 1.183 guidelines to support increasing the unfiltered control room inleakage. The analyses used assumptions consistent with the proposed changes to the PTN Units 3 and 4 licensing basis. With these changes, the resulting calculated doses meet the acceptance criteria as defined in RG 1.183.

RG 1.183 does not address the Spent Fuel Cask Drop or the WGDT Rupture Events. For the Spent Fuel Cask Drop event, the FHA dose limits in RG 1.183 have been adopted. For the WGDT Rupture event, the Branch Technical Position 11-5 of the SRP gives dose acceptance criteria for the exclusion area boundary (EAB) to 100 mrem. RIS 2006-04 establishes the AST dose limit to a member of the public for this event at 100 mrem TEDE. Since RIS 2006-04 associates this acceptance criteria with the annual limit established in 10 CFR 20, the EAB dose will be evaluated over the duration of the event rather than for the worst two-hour period. Branch Technical Position 11-5 does not require the dose consequences to be evaluated at the LPZ or the control room. Although no limit is given to the low population zone (LPZ), the LPZ is evaluated against the 0.1 rem TEDE limit. A TEDE limit of 5.0 rem stated in 10 CFR 50.67 is applied to the control room consistent with other AST events. All doses are evaluated for a 30-day period.

Following the Three Mile Island (TMI) accident, additional evaluations were performed to evaluate the control room to assure operators would be adequately protected. In response to NUREG-0737, "Clarification of TMI Action Plan Requirements" Item II.B.2, FPL installed additional shielding to reduce containment shine radiation exposure from the Unit 3 Purge Valve Exhaust to the control room (Reference 31). Based on this response, the NRC concluded that the modifications were acceptable in accordance with the action plan (Reference 32). As such, the direct shine calculated as part of the AST analysis dose also includes a contribution from the Unit 3 Purge Valves.

Attachment 3, Section 1.6.4 discusses the LOCA shine dose. The 30-day direct shine dose to personnel in the control room, considering occupancy is provided in Attachment 3, Table 1.6.3-2. The total LOCA direct shine dose is 0.723 rem, which is a significant contributor to the total control room LOCA dose consequences (4.87 rem) cited in Attachment 3, Table 2.1-6. Direct shine dose is determined from four different sources to the control room operator after a postulated LOCA event. These sources are containment, containment shine through the Unit 3 purge valves, control room recirculation filters, and the external cloud that envelops the control room. All other sources of direct shine dose are considered negligible. Direct shine dose was calculated using the NRC-approved MicroShield 5 computer code. The applicable components of the LOCA shine dose are used as a conservative assessment of the direct shine dose contribution for all other accidents.

Results of the PTN Units 3 and 4 radiological consequence analyses using the AST methodology and the corresponding allowable control room unfiltered inleakage are summarized in Table 1 below. The analyses support a revised design basis maximum allowable control room unfiltered inleakage of 115 cfm, inclusive of 10 cfm allowance for control room ingress and egress.

Table 1
Summary of AST Analysis Results

Case	Unfiltered Control Room Inleakage (cfm)	Exclusion Area Boundary Dose(1) (rem TEDE)	Low Population Zone Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
LOCA	115	5.85	1.58	4.87
MSLB Pre-accident Iodine Spike	115	0.03	0.02	1.53
SGTR Pre-accident Iodine Spike	115	0.82	0.18	2.85
Acceptance Criteria		**≤ 25**(3)	**≤ 25**(3)	**≤ 5**(4)
MSLB Concurrent Iodine Spike	115	0.05	0.04	1.57
SGTR Concurrent Iodine Spike	115	0.28	0.07	1.10
Locked Rotor	115	0.58	0.62	1.33
Acceptance Criteria		**≤ 2.5** (3)	**≤ 2.5** (3)	**≤ 5** (4)
FHA in Containment	115	0.91	0.20	1.32
FHA in Fuel Handling Building	115	0.91	0.20	3.80
RCCA Ejection – Containment Release	115	0.88	0.40	2.47
RCCA Ejection – Secondary Release (Automatic Control Room Isolation)	115	0.61	0.57	1.19
RCCA Ejection – Secondary Release (Manual Control Room Isolation)	115	0.38	0.36	3.46
Spent Fuel Cask Drop* (with overall DF of 200)	115	0.40	0.09	2.05
Acceptance Criteria		**≤ 6.3**(3)	**≤ 6.3**(3)	**≤ 5**(4)
Waste Gas Decay Tank*	115	0.08	0.02	0.33
Acceptance Criteria		**≤ 0.1**(5)	**≤ 0.1**(5)	**≤ 5**(4)

(1) Worst 2-hour dose applying worst case two hour X/Q
(2) Integrated 30-day dose (Refer to Attachment 3, Section 2.1.5)
(3) Accident Dose Criteria from RG 1.183, Table 6
(4) 10 CFR 50.67
(5) SRP BTP 11-5
* see appropriate event summary in Reference 9 for bases of acceptance criteria

4.2 Changes to the PTN Technical Specifications

The PTN Units 3 and 4 current licensing basis will be revised to fully implement AST, as described in RG 1.183, through reanalysis of the radiological consequences of the UFSAR

Chapters 11 and 14 accidents listed previously. Attachment 3 Section 1.5 describes the Computer Codes used in the AST analyses and Section 1.6 describes the radiological evaluation methodology.

RG 1.183, Regulatory Position 6 states that the NRC Staff is assessing the effects of increased cesium releases on Environmental Qualification (EQ) doses to determine whether licensee action is warranted. NUREG-0933, "Resolution of Generic Safety Issues" (Reference 30) Section 3.0 Item 187 resolved the issues related to the effect of increased cesium releases on EQ doses. In this document, the staff concluded that there was no discernable risk reduction associated adopting the AST for EQ. PTN Units 3 and 4 EQ analyses will continue to be based on TID-14844. Vital Area Access dose and shielding calculations are unchanged with the implementation of AST.

4.2.1 TS 1.12 Definition of Dose Equivalent Iodine

Definition of DOSE EQUIVALENT I-131 is revised to reference EPA FGR 11, as the source of thyroid dose conversion factors.

Current TS

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

Proposed TS

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion."

Justification

A change to the definition of DEI-131 in TS 1.12 is proposed to reference EPA FGR 11 as the source of thyroid dose conversion factors. This change is consistent with the dose conversion factors used in the applicable dose consequence analyses. A precedent for using dose conversion factors from EPA FGR 11 is established in the St. Lucie Unit 1 LAR and Amendment (References 14 and 15, respectively).

4.2.2 TS 1.13 Definition of E-Average Disintegration Energy

The definition of E-AVERAGE DISINTEGRATION ENERGY is deleted and replaced with the definition of DOSE EQUIVALENT XE-133 that includes a reference to EPA FGR 12 as the source of whole body dose conversion factors.

Current TS

1.13 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.

Proposed TS

DOSE EQUIVALENT XE-133

1.13 DOSE EQUIVALENT XE-133 shall be that concentration of XE-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors of air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

Justification

The new definition for DOSE EQUIVALENT XE-133 is similar to the definition for DEI-131. The determination of DOSE EQUIVALENT XE-133 will be performed in a similar manner to that currently used in determining DEI-131, except that the calculation of DOSE EQUIVALENT XE-133 is based on the acute dose to the whole body and considers the noble gases Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 which are significant in terms of contribution to whole body dose. Some noble gas isotopes are not included due to low concentration, short half life, or small dose conversion factor.

When Ē is determined using a design basis approach in which it is assumed that 1.0% of the power is being generated by fuel rods having cladding defects and it is also assumed that there is no removal of fission gases from the letdown flow, the value of Ē is dominated by Xe-133. The other nuclides have relatively small contributions. However, during normal plant operation, there are typically only a small amount of fuel clad defects and the radioactive nuclide inventory can become dominated by tritium and corrosion and/or activation products, resulting in the determination of a value of Ē that is very different than what would be calculated using the design basis approach. Because of this difference, the accident dose analyses become disconnected from plant operation and the Limiting Condition for Operation (LCO) becomes essentially meaningless. It also results in a TS limit that can vary during operation as different values for Ē are determined.

The current Ē definition includes radioisotopes that decay by the emission of both gamma and beta radiation. This change will implement a LCO that is consistent with the whole body radiological consequence analyses which are sensitive to the noble gas activity in the primary coolant, but not to other non-gaseous activity currently captured in the Ē definition. LCO 3.4.8 specifies the limit for primary coolant gross specific activity as 100/Ē microcuries per gram.

This change incorporating the new definition of DOSE EQUIVALENT XE-133 in TS 1.13 using the EPA FGR 12 as the source of whole body dose conversion factors is acceptable from a radiological dose perspective since it will result in an LCO that more closely relates the non-iodine RCS activity limits to the dose consequence analyses which form their bases. This change is consistent with the dose conversion factors used in the applicable dose consequence analyses.

4.2.3 TS 3/4.4.8 RCS Specific Activity

RCS specific activity limit for DEI-131 is being reduced from "less than or equal to 1.0 microcurie per gram" to "less than or equal to 0.25 microcuries per gram" and is being changed from "less than or equal to 100/Ē microcuries per gram of gross radioactivity" to "less than or equal to 447.7 microcuries per gram DOSE EQUIVALENT XE-133." The changes to the LCO, ACTIONS, and SURVEILLANCE REQUIREMENTS are described below:

Current TS

LCO 3.4.8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie per gram DOSE EQUIVALENT I-131, and
b. Less than or equal to 100/Ē microcuries per gram of gross radioactivity.

APPLICABILITY: Modes 1, 2, 3, 4, and 5

ACTION: MODES 1, 2, and 3*:

a. With the specific activity of the reactor coolant greater than 1 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown in Figure 3.4-1, be in at least HOT STANDBY with average reactor coolant temperature less than 500°F within 6 hours; and
b. With the specific activity of the reactor coolant greater than 100/Ē microcurie per gram, be in at least HOT STANDBY with average reactor coolant temperature less than 500°F within 6 hours.

ACTION: MODES 1, 2, 3 4, and 5:

With the specific activity of the reactor coolant greater than 1 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries per gram, perform the sampling and analysis requirements of Item 6.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

* With the average reactor coolant temperature greater than or equal to 500°F.

FIGURE 3.4.-1 DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 1μCi/gram DOSE EQUIVALENT I-131.

TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

Item 1 - Gross Radioactivity Determination – At least once per 72 hours – MODES 1, 2, 3, 4
Item 3 - Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration – 1 per 14 days – MODE 1
Item 5 Radiochemical for Ē Determination – 1 per 6 months* - Mode 1
Item 6 Isotopic Analysis for Iodine Including I-131, I-133, and I-135
Sample And Analysis Frequency –
a) Once per 4 hours whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT I-131 or 100/Ē μCi/gram of gross radioactivity, Modes 1#, 2#, 3#, 4#, 5# and
b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period. Modes 1, 2, 3

TABLE NOTATIONS

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.
# Until the specific activity of the Reactor Coolant System is restored within its limits.

Proposed TS

LCO 3.4.8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.25 microcuries per gram DOSE EQUIVALENT I-131, and
b. Less than or equal to 447.7 microcuries per gram DOSE EQUIVALENT XE-133.

APPLICABILITY: MODES 1, 2, 3, and 4:

ACTION:

a. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is less than or equal to 60 microcuries per gram once per 4 hours.

b. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 60 microcuries per gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 0.25 microcuries per gram limit. Specification 3.0.4 is not applicable.

c. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131 for greater than or equal to 48 hours during one continuous time interval, or greater than 60 microcuries per gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

d. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 447.7 microcuries per gram limit. Specification 3.0.4 is not applicable.

e. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133 for greater than or equal to 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

Page 3/4 4-27
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TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM
Item 1 NOT USED
Item 3 Isotopic Analysis for DOSE EQUIVALENT I-131–

Sample And Analysis Frequency –
a) 1 per 14 days – Mode 1 and
b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period – Mode 1

Item 5 Isotopic Analysis for DOSE EQUIVALENT XE-133 – 1 per 7 days – Mode 1

Item 6 NOT USED

Page 3/4 4-29
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Justification

The proposed limit for DEI-131 is more restrictive than the limit currently in place and consistent with that assumed in the accident dose consequences analysis. The proposed TS limit for DEI-131 is approximately 40 times higher than the highest measured DEI-131 from previous operating cycles that contained fuel failures. Thus, reducing the RCS specific activity limit from 1.0 microcurie per gram to 0.25 microcuries per gram of DEI-131 will not result in an undue burden on plant operation.

TS LCO 3.4.8.b replaces $100/\bar{E}$ with DOSE EQUIVALENT XE-133 less than or equal to 447.7 microcuries per gram. This limit is established based on the RCS activity corresponding to 1% fuel clad defects with sufficient margin to accommodate the exclusion of those isotopes based on low concentration, short half life, or small dose conversion factors and is consistent with that assumed in the accident dose consequences analysis. The primary purpose of the TS 3.4.8 LCO on RCS specific activity and its associated actions is to support the dose analyses for design basis accidents. The whole body dose is primarily dependent on the noble gas activity, not the non-gaseous activity currently captured in the $\bar{E}$ definition.

TS 3.4.8 Applicability removes MODE 5. It is necessary for the LCO to apply during MODES 1 through 4 to limit the potential radiological consequences of a SGTR or MSLB that may occur during these MODES. In MODE 5 with the RCS loops filled, the SGs are specified as a backup means of decay heat removal via natural circulation. In this mode, however, due to the reduced temperature of the RCS, the probability of a design basis accident involving the release of significant quantities of RCS inventory is greatly reduced. Therefore, monitoring of RCS specific activity is not required. In MODE 5 with the RCS loops not filled, the SGs are not used for decay heat removal; the RCS and SGs are depressurized and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required. The change to modify the TS 3.4.8 Applicability to include only MODES 1 through 4 retains the necessary constraints to limit the potential radiological consequences of a SGTR or MSLB that may occur during these MODES and is therefore acceptable from a radiological dose perspective.

Figure 3.4.-1, DEI-131 Reactor Coolant Specific Activity Limit vs. Rated Thermal Power with the Reactor Coolant Specific Activity > 1µci/gram DEI-131, is deleted. The new limit is 60 microcuries per gram DEI-131 at all power levels. Action a. has been changed to delete the reference to Figure 3.4.-1 and to state greater than 60 microcuries per gram DEI-131. The change from a graph that changes with power level to a specific value is consistent with dose consequence events that are analyzed at full-power and assume a pre-accident spike of 60 microcuries per gram DEI-131. The full power transients that allow a DEI-131 spike are not changed.

New Actions a. through e. are incorporated to implement the new LCOs for DEI-131 and DOSE EQUIVALENT XE-133 and to replace Figure 3.4-1 with a specific value for DEI-131 of 60 microcuries per gram. Actions a. through c. provide the requirements for DEI-131. Actions d. and e., provide the requirements for DOSE EQUIVALENT XE-133. The required actions and completion times are consistent with the required actions and completion times in the current TS. However, Actions c. and e. have been revised to include a requirement to be in COLD SHUTDOWN within the next 30 hours. This requirement will ensure the unit is placed in a Mode where the TS is not Applicable.

Actions b. and d. are revised to state that LCO 3.0.4. is not applicable. This will allow entry into a Mode or other specified condition in the LCO Applicability when LCO 3.4.8 is not being met. The proposed change to Action b. would allow entry into the applicable Modes from MODE 4 (HOT SHUTDOWN) to MODE 1 (POWER OPERATION) while the DEI-131 is greater that the 0.25 microcuries per gram and less than or equal to 60 microcuries per gram and DEI-131 is being restored to within its limit. The change to Action d. would allow entry into the applicable Modes from MODE 4 (HOT SHUTDOWN) to MODE 1 (POWER OPERATION) while the DOSE EQUIVALENT XE-133 is greater than 447.7 microcuries per gram and the DOSE EQUIVALENT XE-133 is being restored to within its limit. This Mode change is acceptable due to the significant conservatism incorporated into the DEI-131 and DOSE EQUIVALENT XE-133 specific activity limits, the low probability of an event occurring which is limiting due to exceeding the specific activity limits, and the ability to restore transient specific excursions while the plant remains at, or proceeds to power operation.

TS Table 4.4-4, Item 5 is changed from "Radiochemical for $\bar{E}$ Determination" to "Isotopic Analysis for DOSE EQUIVALENT XE-133." This surveillance requires a gamma isotopic analysis as a measure of the noble gas activity of the reactor coolant at least once every 7 days. The measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. The surveillance provides an indication of any increase in noble gas specific activity. Trending the results of this surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The surveillance frequency of 7 days considers the low probability of a gross fuel failure during this time. Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the surveillance calculation. If a specific noble gas nuclide listed in the definition for DOSE EQUIVALENT XE-133 is not detected, it will be assumed to be present at the minimum detectable activity. The Table Notation "*" is deleted. This is an administrative change because the notation was only applicable to the Ē Determination which has been replaced with DOSE EQUIVALENT XE-133. TS Table 4.4-4, Item 1 "Gross Radioactivity Determination" is deleted since the measurement of gross radioactivity applied to the Ē Determination which has been replaced with DOSE EQUIVALENT XE-133. This is also an administrative change to conform with a DOSE EQUIVALENT XE-133 limit.

TS Table 4.4-4 Item 3 is revised to delete the word "concentration." This is an administrative change. The surveillance ensures that the specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14 day frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Mode Applicability and frequency are unchanged from the current TS for Item 3 a). The frequency of between 2 and 6 hours after a power change greater than or equal to 15% rated thermal power within a 1 hour period is established because the iodine levels peak during this time following fuel failure; samples at other frequencies would provide inaccurate results. The sample frequency remains unchanged. The Mode Applicability is appropriate since there is a low probability of fuel failure in Modes 2 and 3.

TS Table 4.4-4, Item 6 "Isotopic Analysis for Iodine Including I-131, I-133, and I-135" is deleted since this is redundant to Action a. Likewise, the sampling requirement following a 15% rated thermal power change is included in Item 3. The Table Notation "#" is deleted. This is an administrative change because the notation was only applicable to Item 6 which has been deleted.

The proposed TS changes for Reactor Coolant Specific Activity have been previously approved for Millstone Power Station Units 2 and 3 by Reference 16.

4.2.4 TS 3/4.6.3 Emergency Containment Filtering System

The Emergency Containment Filtering (ECF) System TS is deleted and replaced with Not Used.

Justification

The ECF System is credited in the current PTN accident analyses for dose consequences. The ECF system is not credited in the AST analyses and no longer required to mitigate any design basis accident. ECF System iodine removal capability is no longer required since the dose analyses are within regulatory limits. As such, the ECF system does not meet the 10 CFR 50.36 criteria to for inclusion in the TS. Credit for the Emergency Containment Filtering System will be removed from the PTN Units

3 and 4 licensing basis. The proposed change in this LAR will delete TS 3/4.6.3, ECF System.

The ECF System changes will not have a significant impact on the PTN PRA Level 2 analysis, since the ECFs are not modeled in the PTN PRA. The Level 2 PRA results are focused on calculating Large Early Release Frequency (LERF). LERF is dominated by bypass sequences, such as core damage sequences initiated by interfacing system LOCAs or SGTRs.

The ECF System is a feature in the Severe Accident Mitigation Guidelines (SAMG). Since the calculated doses are within regulatory limits, removal of this system does not create an adverse impact on the severe accident management program.

4.2.5 TS 3/4.7.5 Control Room Emergency Ventilation System

Control Room Emergency Ventilation System Surveillance Requirement 4.7.5.c 2, is changed from "methyl iodide penetration criteria of less than 2.5%" to "methyl iodide penetration criteria of less than 1.25%."

Current TS

SR 4.7.5.c.2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ASTM D3803 – 1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 2.5% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement, and

Proposed TS

SR 4.7.5.c.2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ASTM D3803 – 1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 1.25% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement, and

Justification

TS 4.7.5.c.2 requires a change from "methyl iodide penetration criteria of less than 2.5%" to "methyl iodide penetration criteria of less than 1.25%." The analyses assume a Control Room Emergency Ventilation Filtration organic removal efficiency of 97.5%. The TS is being reduced to a methyl iodide penetration criteria of 1.25% to ensure that a minimum safety factor of 2 is maintained as required by GL 99-02, "Laboratory Testing of Nuclear- Grade Activated Charcoal" (Reference 17). PTN submitted a LAR and received TS amendments 205/199 (Reference 18) to address the Control Room

Emergency Ventilation Systems. The safety evaluation report for the TS amendments required a minimum safety factor of 2 for iodine penetration. This change will ensure that the requirements of GL 99-02 and TS amendments 205/199 safety factors are maintained. Recent TS surveillance results have shown that methyl iodide penetration is historically less than 1.25%, such that reducing the methyl iodide penetration criteria from less than 2.5% to methyl iodide penetration criteria less than 1.25% will not result in a burden on plant operation.

4.2.6 TS 3.7.9 Gas Decay Tanks

Gas Decay Tanks LCO is clarified to change the phrase within the parentheses from "(considered as Xe-133 equivalent)" to "(DOSE EQUIVALENT XE-133)".

Current TS

The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 Curies of noble gases (considered as Xe-133 equivalent).

Proposed TS

The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 Curies of noble gases (DOSE EQUIVALENT XE-133).

Justification

The terminology used in LCO 3.7.9 to establish the maximum curie content for noble gases is changed from "considered as Xe-133 equivalent" to "DOSE EQUIVALENT XE-133." This is an administrative change to provide clarity. The maximum curie content limit remains unchanged.

4.2.7 TS 6.8.4h. Containment Leakage Rate Testing Program

A proposed change to the TS maximum allowable containment leakage limit in TS 6.8.4.h, will reduce the L_a limit from 0.25% to 0.20% of containment air weight per day.

Current TS

The maximum allowable containment leakage rate, La at Pa, shall be 0.25% of containment air weight per day.

Proposed TS

The maximum allowable containment leakage rate, La at Pa, shall be 0.20% of containment air weight per day.

Justification

The proposed limit is more restrictive than the limit currently in place. The maximum allowable leakage rate L_a of 0.20% supports the value assumed in the design basis LOCA and the RCCA Ejection event analyses to determine the dose consequences. Attachment 3 Section 2.1 discusses the radiological consequences of a design basis

LOCA and Section 2.6 discusses the radiological consequences of the RCCA Ejection event.

The result of the most recent containment Integrated Leakage Rate Tests (ILRT) for PTN Units 3 and 4 were leak rates of 0.08% and 0.11% of containment air weight per day respectively, which is significantly less than the acceptance limit criterion of 0.1875%. The acceptance criterion is defined as 0.75 L_a, where in this case, L_a is equal to 0.25% of the containment air weight per day. By reducing the TS L_a value to 0.20% of the containment air weight per day, the acceptance limit criterion now becomes 0.15%. Given the new acceptance criterion of 0.15%, the actual tested leak rates of 0.08% and 0.11% are still below the limit. Additionally, the PTN Units 3 and 4 ILRT As-Found Leak Rate data from 1982 to 1992 resulted in leak rates between 0.05% and 0.11%. A review of the historical data from ILRT and an engineering evaluation validates the data and supports the change in the allowable limit acceptance criterion.

4.2.8 TS 6.8.4.j.b.2, Steam Generator Program

The SG Program is revised to reduce the accident-induced primary to secondary leakage performance criterion from one gpm through all SGs and 500 gallons per day (gpd) through one SG at accident conditions to 0.20 gpm through any one SG and 0.60 gpm through all SGs at room temperature conditions (approximately 398 gpd at accident conditions).

Current TS

Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total through all SGs and 500 gallons per day through any one SG.

Proposed TS

Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions.

Justification

The accident induced leakage performance criterion for primary to secondary leakage is being reduced as described above. The proposed limits are lower than the limits currently in place are consistent with the leakage rates assumed in the accident analyses, other than SGTR, in terms of total leakage rate for all SGs and leakage rate for an individual SG. The operational limit on primary to secondary leakage limit in any one SG is 150 gpd at room temperature. Therefore, there is approximately a 2 to 1 margin between the operational and the accident induced primary to secondary leakage limits.

TS 3.4.6.2, RCS Operational Leakage, defines the limit for primary to secondary leakage through any one SG as 150 gpd and is unchanged by the AST analyses. This will have no impact on plant operations. The AST analyses conservatively assume the primary to secondary leakage to be maximized at 0.20 gpm using a room temperature water density.

4.2.9 TS 6.9.1.2 Annual Reports

Annual report for specific activity analysis is changed to address the changes in TS 3.4.8 RCS specific activity discussed above. The footnote ** at the bottom of the page is deleted.

Current TS

(5) The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie per gram DOSE EQUIVALENT I-131.

** This tabulation supplements the requirements of §20.2206 of 10 CFR Part 20.

Proposed TS

(5) The time duration when the specific activity of the primary coolant exceeded 0.25 microcuries per gram DOSE EQUIVALENT I-131.

** The footnote is deleted.

Justification

This is an administrative change to ensure that the reporting requirements agree with the applicable TS LCO and ACTION requirements and to delete an extraneous footnote. There is no text on the page that requires this footnote.

4.2.10 TS 3/4.6.2.3 Recirculation pH Control System

This is a new TS to control the pH of the post-LOCA containment sump solution using NaTB. This passive system will consist of ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the lower regions of the containment.

Proposed TS

See Attachment 1, Marked Up Technical Specification Pages, Insert - page 3/4 6-15.

Justification

RG 1.183 Appendix A Regulatory Position 2 requires the recirculation sump pH be controlled at values of 7 or greater at the onset of containment spray recirculation mode. To achieve a pH of 7 at the onset of recirculation, NaTB basket will be used to neutralize the containment sump water. This passive system will consist of ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the lower regions of the containment.

As stated in WCAP-16596-NP (Reference 19), a fully-hydrated form of NaTB is proposed which makes it less likely that large amounts of water will be absorbed from the potentially humid containment conditions during plant operation. If exposed to dry containment conditions, the hydrated NaTB has the potential for some loss of water. The chemical properties of the buffer do not change as a result of the potential water loss, but the weight does decrease. The use of NaTB for post-LOCA containment sump pH control has been previously approved by the NRC in Indian Point 3 license amendment 224 (Reference 20).

FPL has performed an analysis to determine the amount of NaTB needed to maintain the post-LOCA containment sump pH greater than or equal to 7.0 following recirculation at PTN. The analysis used dissolution data contained in WCAP-16596-NP to calculate surface dissolution rates. The analysis considered the minimum and maximum quantities of boron and borated water, the time-dependent post-LOCA sump temperatures, and the minimum time to reach switchover conditions. In addition, the formation of acid from radiolysis of air and water, radiolysis of chloride bearing electrical cable insulation and jacketing, and spilled reactor coolant were considered.

The analysis determined acceptable basket configurations for obtaining a sump pH of 7.0 at the onset of the containment spray recirculation mode. The results showed that the minimum number of baskets would be two large baskets and eight small baskets. Using this basket configuration, a parametric study was performed to determine the acceptable NaTB level required to ensure a pH of 7.0 was achieved at the onset of the containment spray recirculation mode. Since, not all of the NaTB will be dissolved at the onset of containment spray recirculation, the final long-term pH was also calculated. Table 2 provides a summary of the quantities of NaTB and resulting pH values at the onset of recirculation and long-term.

The analysis performed parametric studies to determine acceptable basket configurations that ensure a sump pH of 7.0 at the minimum time to initiate containment recirculation spray flow. The results of Case 2, provided in Table 2 below, show the minimum quantity of NaTB required. Case 2 conservatively assumed the following:

- Minimum NaTB density – This maximizes the required number of baskets and dissolution time.
- Maximum refueling water storage tank (RWST), accumulator, and RCS inventory and boric acid concentration – This maximizes the NaTB requirement and minimizes the sump pH.
- Maximum chloride bearing electrical cable insulation and jacketing mass – This maximizes the quantity of strong acid generation by radiolysis, thereby, minimizing the sump pH.
- Minimum sump level versus time profile – The analysis conservatively adjusted the sump level versus time profile, such that the sump level is minimized at the earliest onset of containment spray recirculation flow, thereby maximizing the

concentrations of boric acid and strong acids in the sump.

The results of Case 3, provided in Table 2, shows that the maximum 30-day pH remains less than 8.2, which is the long-term maximum pH discussed in the following paragraphs. This case conservatively assumes the following:

- Maximum NaTB density;
- Minimum boric acid concentrations in the RCS and RWST;
- No strong acids considered;
- Maximum sump temperature;
- Maximum RCS inventory in the sump – This is conservative since the assumed boric acid concentration for this case is 0 ppm;
- No accumulator volume is considered;
- Minimum RWST level;
- All baskets are completely full of NaTB;
- One additional small basket completely full of NaTB is installed in containment;

The concentration of hydrogen ions is based on the relative level of acids and bases and other ions in the sump solution. For evaluating the post-LOCA sump pH, the following chemical compounds are considered:

- Boron/boric acid
- NaTB
- Hydrochloric acid
- Nitric acid

Hydrochloric acid generated as a result of the irradiation of chloride bearing cable insulation and jacketing in containment and nitric acid generated by irradiation of water in the sump are determined following the guidelines of NUREG/CR-5950. The quantities of hydrochloric acid and nitric acid are determined based on the 30-day integrated post-LOCA gamma and beta doses in the containment air and sump. The minor contributions from other acidic and basic species are assumed to offset each other and are negligible in comparison to the chemicals above.

TABLE 2							
Case No.	Purpose Min/Max pH	Small Basket Height (ft)	Large Basket Height (ft)	Initial NaTB Mass (lbm)	Dissolved Mass prior to Recirc (lbm)	pH at Recirc	Long-Term pH
1	Min	2.5	2.77	14264	4687	7.005	7.396 (min)
2	Min	1.9167	2.1867	11061	4637	7.000	7.241 (min)
3	Max	2.5	2.77	17034*	17034	8.105	8.108 (max)

*This case conservatively assumed that an additional small basket is installed in containment.

The baskets containing the NaTB will facilitate a passive mechanism for controlling post-accident containment sump fluid pH without operator action. The two large baskets and eight small baskets are designed to contain a combined mass of greater than 11,061 lbm of NaTB, thereby ensuring that the minimum pH of the sump at the onset of spray recirculation flow is 7.0. The above quantity of NaTB will ensure that the post-LOCA containment sump pH is maintained greater than or equal to 7.0 to support the containment radioiodine specification assumptions in the LOCA radiological consequences analysis detailed in Attachment 3. A proposed change to the TS will incorporate a new TS 3/4.6.2.3, Recirculation pH Control System. This specification will ensure operability of the passive post-LOCA pH control system.

Chloride induced stress corrosion cracking (SCC) of austenitic stainless steel components is reduced if the pH of the sump is maintained greater than 7.0. Short term exposure of less than one hour to a pH less than 7.0 prior to onset of the recirculation phase will not result in significant SCC, but long term exposure of several days during the recirculation phase may result in significant damage. NUREG-0800, SRP Section 6.1.1.1 (Branch Technical Position MTEB 6-1) (Reference 21) recommends that the long term containment sump pH should be maintained between 7.0 to 9.5. PTN EQ Documentation Packages establish the minimum (7.0) and maximum (8.5) pH limits assumed for equipment qualification.

Currently at PTN, the post-LOCA sump pH is controlled by manually adding NaTB via the Chemical Volume and Control System (CVCS) in accordance with the Emergency Operating Procedures (EOPs) until the pH is adjusted above 7.2. The pH analysis shows that the pH will be adjusted above 7.0 very early into the event, well before the recommended time to prevent SCC of stainless steal components. As shown by Case 3, the maximum possible pH is 8.1. This case included an extra small basket of NaTB to demonstrate that a maximum pH value would not approach pH levels which could cause SCC. The proposed recirculation sump pH control system, therefore, will sufficiently prevent degradation of stainless steals in the containment building.

GL 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors," (Reference 22) discusses the potential for larger quantities of debris to be generated during a LOCA and the effect of more debris on mitigating systems. The installation of the ten stainless steel baskets containing NaTB will maintain the sump pH at or above 7.0. To account for minimum and maximum values that influence pH and fixed quantities of NaTB, the maximum pH can not be adjusted to 7.2. Case 3 shows the maximum pH will be increased above the current value. The increased pH will result in the generation of additional debris. Currently, the primary source of chemical debris in containment is from aluminum in the normal containment cooler fins. To offset the effects from the increased pH, the normal containment cooler aluminum fins are being replaced with copper fins which generate less debris. The overall result of increasing the containment sump maximum pH and replacing the aluminum fins with copper fins will maintain the post-LOCA debris generation within the quantities currently reported in the PTN sump strainer design basis. The proposed changes will not increase the potential debris in the containment sump; therefore, PTN will continue to satisfactorily address Generic Safety Issue (GSI) 191, "Assessment of [Effect of] Debris Accumulation on PWR Sump Performance."

NRC Branch Technical Position MTEB 6-1 indicates that: "for a pH above 7.5, consideration should be given to hydrogen generation problems arising from corrosion of aluminum." As discussed above, the quantity of aluminum within containment will be substantially reduced with the installation of copper fins for the normal containment coolers. Their replacement will act to offset any potential increase in aluminum corrosion. However, the NRC granted approval of PTN license amendments 217/211 (Reference 23) deleting the TS for hydrogen monitors and the associated post-accident containment ventilation system. The NRC safety evaluation concluded that hydrogen combustion is a small contributor to containment failure for large/dry containment designs due to the robustness of these containment types and the probability of common sources of random ignition from electrical equipment sparks and small static electric discharges. The basis for approval of these amendments is not impacted by installation of the NaTB baskets. Since hydrogen generation is not considered a post-LOCA issue for the PTN design, the pH values that will result from implementation of this LAR are acceptable.

Appropriate TS LCO, Actions, and Surveillance Requirements are proposed for TS 3/4.6.2.3 "Containment Systems – Recirculation pH Control System." Containment sump pH control systems meet the 10 CFR 50.36 Criterion 3 threshold for inclusion within TS since they are part of a primary success path to mitigate the consequences of a design basis accident. The proposed TS, LCOs, and Actions are consistent with NUREG-1431, "Standard Technical Specifications – Westinghouse Plants" (Reference 24). NUREG-1431 provide surveillance requirements for an active system. As such, the proposed LCO Applicability, Actions and Completion Times for an inoperable Recirculation pH Control System are consistent with other recently approved Containment Systems – Spray Additive System TS (Reference 26 and 27). The proposed Surveillance Requirements ensure that the total amount of NaTB stored in containment is sufficient to ensure that a sump pH of 7.0 will be achieved by the start of sump recirculation and maintain the post-LOCA sump water pH greater than or equal to 7.0 thereafter.

4.3 Control Room Envelope Inleakage

Control room tracer gas testing per ASTM E-741-2000, "Standard Test Method for Determining Air Change in a Single Zone by Means of Tracer Gas Dilution" (Reference 25) was performed on the control room envelope for PTN Units 3 and 4. The results demonstrated that the control room envelope is at a positive pressure to all adjacent areas and that the outside airflow is the only source of inleakage to the envelope in the emergency pressurization mode and that this entire inleakage is filtered. Specifically, the test determined, within a 95% confidence level, that the average outside air flow exceeded the average total control room inleakage. Therefore, the control room bypass leakage assumed in the AST analyses conservatively bounds the as-tested inleakage.

4.4 Post-LOCA Iodine Removal

As discussed in Attachment 3, Section 2.1.3, reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine has been calculated and is credited for the sprayed and unsprayed regions. Natural deposition for the removal of aerosols is credited for unsprayed regions only. No natural deposition for the removal of organic iodine is credited.

Containment spray provides coverage of a portion (approximately 1/3) of the containment volume. Therefore, the PTN containment atmosphere is not considered to be a single, well-mixed volume. The containment is divided into three regions, sprayed and unsprayed above the operating deck and an unsprayed region below the operating deck. The mixing rates between the regions are based on a separate sensitivity study evaluating various combinations of containment fans and sprays to produce the most conservative mixing rates. Containment spray is credited for the removal of elemental and particulate (aerosols) iodine. The elemental iodine spray removal and the particulate iodine spray removal coefficients are dependent on the calculated decontamination factors (DF), based on the maximum airborne elemental and aerosol concentrations in the containment. Elemental iodine removal capability stops when the DF reaches 200 (approximately 2.3 hours). The spray aerosol removal rate is reduced by a factor of 10 when the DF reaches 50 (approximately 3 hours).

Containment spray flow is automatically initiated post-LOCA following actuation of a containment high pressure signal. EOPs currently direct operators to realign the containment spray pump to take suction from the containment sump via the RHR pump discharge during switchover to cold leg recirculation. Therefore, credit for elemental and aerosol DFs is achievable based on the current plant design.

4.5 Control Room Ventilation System Isolation

Redundant radiation monitors located on the control room normal air intake ensure that the emergency ventilation system has redundant means of automatic actuation for all postulated accidents having the potential of yielding doses greater than GDC 19 limits. To achieve this for the cask drop accident, FPL installed redundant safety related normal air intake radiation monitors to automatically initiate control room isolation/pressurization upon detection of high radiation (Reference 33). The addition of this automatic isolation signal ensured that LOCA remained the most severe design basis accident with regard to control room doses.

For the AST analysis, the Locked Rotor and secondary release RCCA Ejection Events exceed the control room detector setpoint within 30 seconds. An additional 30 seconds account for the signal processing, and damper closure times. TS 3.3.2 Engineered Safety Features Actuation System Instrumentation, Functional Unit 9.e ensures the operability of the control room air intake radiation monitor and associated trip setpoint surveillance criteria are maintained.

For events which do not initiate a control room isolation signal, the analyses credit operator action to initiate control room isolation/pressurization within 30 minutes. For the FHA, Spent Fuel Cask Drop, and secondary release RCCA Ejection with less than design basis fuel failure events, no credit was taken for an automatic control room isolation signal. As such, these analyses credit manual operator action to isolate the control room in 30 minutes. This action can be recognized using existing control room indications from primary and secondary system process radiation monitors and achieved from the control room within the specified time. Procedures exist to respond to these events.

Similarly, Palisades has set precedence by assuming their control room would be manually isolated and placed into emergency recirculation mode in 20 minutes for all analyzed events (Reference 34).

4.6 Disposition of Recent AST Submittals RAIs

A review of NRC Requests for Additional Information (RAIs) received concerning the recent AST submittals was performed. The subjects of the RAIs that could be applicable to PTN Units 3 & 4 have been reviewed and incorporated in this LAR, as appropriate.

5.0 List of Commitments

1. FPL will relocate the Control Room Ventilation System emergency air intakes prior to implementation of AST. The relocated intakes and associated ductwork will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR 50 Appendix A, GDC 19. The new intakes will be located near the ground level of the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for post-accident contaminants. FPL will perform post-modification testing in accordance with the plant design modification procedures to ensure the TS pressurization flow remains adequate to demonstrate the integrity of the relocated intakes.

2. FPL will install ten (two large and eight small) stainless steel wire mesh baskets containing NaTB located in the containment basement to maintain pH during the sump recirculation phase following a Design Basis LOCA.

3. FPL will implement the requirements of the proposed Technical Specification for the Recirculation pH Control System to maintain operability of this system and ensure the suitability of the NaTB. This commitment supersedes the commitment from PTN Letters L-2009-062 and L-2009-063. (References 28 and 29, respectively) related to NaTB sampling.

4. FPL will replace the aluminum fins on the normal containment coolers with copper fins. The copper fins will maintain the post-LOCA debris generation within the quantities currently assumed in the PTN sump strainer design basis.

5. FPL will update the necessary procedures to implement the manual operator action for initiation of the emergency control room ventilation system.

6.0 Conclusion

The use of the AST methodology in accordance with this LAR for the design basis accident dose consequences analyses will support increases in allowable control room unfiltered inleakage. The analyses performed in support of this LAR meet all of the applicable criteria of 10 CFR 50.67, as defined within RG 1.183.

This LAR was developed using the guidance provided in RIS 2006-04. The proposed new TS for Section 3/4.6 "Containment Systems" ensures that the containment sump pH is maintained at or above 7.0 in accordance with the LOCA AST radiological consequence analysis.

7.0 No Significant Hazards Determination

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazard if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Florida Power and Light (FPL) proposes to revise the Turkey Point Units 3 and 4 licensing basis to implement Alternative Source Term (AST), described in Regulatory Guide 1.183 (July 2000). The following limiting design bases accidents are analyzed:

- Loss-of-Coolant Accident
- Main Steam Line Break
- Steam Generator Tube Rupture
- Locked Rotor
- Rod Cluster Control Assembly Ejection
- Fuel Handling Accident
- Spent Fuel Cask Drop, and
- Waste Gas Decay Tank Rupture

NRC Regulatory Guide (RG) 1.183 does not address the Spent Fuel Cask Drop and the Waste Gas Decay Tank Rupture events. For the Spent Fuel Cask Drop event, the requirements of RG 1.183 for the Fuel Handling Accident have been adopted. For the Waste Gas Decay Tank Rupture event, the limit of 0.1 rem total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) established in RIS 2006-04 has been adopted. Although no limit is given to the low population zone (LPZ), the LPZ is also evaluated against the 0.1 rem TEDE limit. A TEDE limit of 5.0 rem stated in 10 CFR 50.67 is applied to the control room. All doses are evaluated for a 30-day period.

FPL has reviewed this proposed license amendment for FPL's Turkey Point Units 3 and 4 and determined that its adoption would not involve a significant hazards consideration. The bases for this determination are:

The proposed amendment does not involve a significant hazards consideration for the following reasons:

1. **The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   AST calculations have been performed for Turkey Point Units 3 and 4 which demonstrate that the dose consequences remain below limits specified in RG 1.183 and 10 CFR 50.67. For the Spent Fuel Cask Drop and the Waste Gas Decay Tank Rupture Events which are not addressed by RG 1.183, the AST methodology has demonstrated that the dose consequences remain below the limits identified above. The AST calculations are based on the current plant design and operation as modified by the installation of a passive post-LOCA recirculation pH control system, re-location and redesign of the control room emergency ventilation intakes, the replacement of the aluminum normal containment cooler fins with copper fins, and for certain events, manual operator actions for initiation of control room emergency ventilation system. These proposed changes to the plant configuration are not accident precursors for any previously evaluated accidents and support mitigation of the dose consequences of previously evaluated accidents. The proposed modification to the plant configuration will be fully qualified to the appropriate design requirements to assure their required function is available for accident mitigation and to assure the function of other equipment required for accident mitigation are not adversely impacted. The use of the AST changes the regulatory assumptions regarding the analytical treatment of the design basis accidents and has no direct effect on the probability of any accident. The AST has been utilized in the analysis of the limiting design basis accidents listed above. The results of the analyses, which include the proposed changes to the Technical Specifications (TS), and the installation of the modifications, demonstrate that the dose consequences of these limiting events are all within regulatory limits.

TS 3/4.6.3 Emergency Containment Filtering (ECF) System has been deleted since the dose consequence analyses are within regulatory limits. A new TS is being incorporated to ensure the operability of the Recirculation pH Control System. The remaining TS changes are consistent with, or more restrictive than, the current TS requirements or established precedent. None of the affected systems, components, or programs are related to accident initiators.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed changes to Turkey Point Units 3 and 4 only affect those systems described above. The proposed Recirculation pH Control System is a passive system that will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed modification to the plant configuration will be fully qualified to the appropriate design requirements to assure their required function is available for accident mitigation and to assure the function of other equipment required for accident mitigation are not adversely impacted. Neither implementation of the AST methodology, establishing more restrictive TS requirements, deleting TS 3/4.6.3, nor installing the modifications described above have the capability to introduce any new failure mechanisms or cause any analyzed accident to progress in a different manner.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed amendment does not involve a significant reduction in the margin of safety.**

    The proposed implementation of the AST methodology is consistent with NRC Regulatory Guide 1.183. For the Spent Fuel Cask Drop and the Waste Gas Decay Tank Rupture Events which are not addressed by RG 1.183, the AST methodology has demonstrated that the dose consequences remain below the limits identified above.

    With the exception of the deletion of TS 3/4.6.3, and the addition of the recirculation pH sump control system, the proposed TS changes are consistent with, or more restrictive than, the current TS requirements or established precedent. The proposed TS requirements and plant modifications will support the AST revisions to the limiting design basis accidents. As such, the current plant margin of safety is preserved. Conservative methodologies, per the guidance of RG 1.183, have been used in performing the accident analyses. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with the use of the AST methodology.

    The proposed changes continue to ensure that the doses at the exclusion area and low population zone boundaries and in the Control Room are within the corresponding regulatory limits of RG 1.183 and 10 CFR 50.67. The margin of safety for the radiological limits is set at or below the 10 CFR 50.67 limits. An acceptable margin of safety is inherent in these limits.

    Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above discussion, FPL has determined that the proposed change does not involve a significant hazards consideration.

8.0 Environmental Consideration

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment of an operating license for a facility requires no environmental assessment, if the operation of the facility in accordance with the proposed amendment does not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (3) result in a significant increase in individual or cumulative occupational radiation exposure. FPL has reviewed this LAR and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment. The basis for this determination follows.

Basis

This change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in the 10 CFR 50.92 evaluation, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite. The change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released offsite.

3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed change is mainly analytical. The addition of stainless steel wire mesh baskets inside containment has no impact on the cumulative occupational radiation exposure. The surveillance of the amount of NaTB in the baskets will be done during a refueling outage so as to ensure that no significant dose is received. Additionally, trained operators following procedures will respond to the individual events to mitigate radiation exposures. Therefore, the proposed amendment has no significant affect on either individual or cumulative occupational radiation exposure.

9.0 Summary of Results

Results of the PTN radiological consequence analyses using the AST methodology and the corresponding allowable control room inleakage are summarized on Table 1 in Section 4.1.5.

The total amount of NaTB stored in containment is sufficient to ensure that a sump pH of 7.0 will be achieved by the start of sump recirculation and maintain the post-LOCA sump water pH greater than or equal to 7.0, thereafter. A discussion of the containment sump pH analysis is provided in Section 4.2.10.

10.0 References

1. 10 CFR, Code of Federal Regulations, Title 10 Energy.

2. Turkey Point Units 3 and 4, Updated Final Safety Analysis Report (UFSAR).

3. TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, March 23, 1962.

4. USNRC, Regulatory Guide (RG) 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

5. Regulatory Information Summary (RIS) 2006-04, Experience with Implementation of Alternative Source Terms, May 7, 2006.

6. NEI 99-03, Control Room Habitability Guidance, Nuclear Energy Institute, Revision 0, June 2001 and Revision 1, March 2003.

7. Generic Letter 2003-01, Control Room Habitability, June 12, 2003.

8. J. A. Stall (FPL) Letter (L-2009-299) to USNRC Document Control Desk, Generic Letter 2003-01, Control Room Habitability – 180 Day Response, December 9, 2003.

9. NAI-1396-045, AST Licensing Technical Report for Turkey Point Units 3 and 4, Revision 1, Numerical Applications, Inc., June 18, 2009.

10. EPA Federal Guidance Report No. 11 (FGR 11), Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, 1988.

11. EPA Federal Guidance Report No. 12 (FGR 12), External Exposure to Radionuclides in Air, Water, and Soil, 1993.

12. USNRC, Regulatory Guide (RG) 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003.

13. Regulatory Guide (RG) 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, November 1982.

14. G.L. Johnston (FPL) to NRC Document Control Desk (L-2007-085), License Amendment Request "Alternative Source Term and Conforming Amendment," July 16, 2007.

15. B.L. Mozafari (NRC) to J.A. Stall (FPL), St. Lucie Plant, Unit 1 - Issuance Of Amendment Regarding Alternative Source Term, November 26, 2008.

16. C. Sanders (NRC) to D.A. Christian (Virginia Electric and Power Company), "Millstone Power Station , Units 2 and 3 – Issuance of Amendment Re: Deletion of E Bar Definition and Revision to the Reactor Coolant System Specific Activity (TAC Nos. MD8492 and MD8493)," October 27, 2008.

17. Generic Letter (GL) 99-02, Laboratory Testing of Nuclear Grade Activated Charcoal, June 3, 1999.

18. Safety Evaluation By The Office Of Nuclear Reactor Regulation Related to Amendment No. 205 to Facility Operating License No. DPR-31 and Amendment No. 199 to Facility Operating License No. DPR-41, March 21, 2000.

19. WCAP-16596-NP, "Evaluation of Alternative Emergency Core Cooling System Buffering Agents," Revision 0, July 2006.

20. J.P. Boska (NRC) To Vice-President, Entergy Nuclear Operations, "Indian Point Nuclear Generating Unit No. 3 - Issuance Of Amendment Re: Changes to Technical Specifications to Replace Sodium Hydroxide Buffer with Sodium Tetraborate," June 9, 2008.

21. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 6.1.1, Engineered Safety Features Materials.

22. GL 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During design Basis Accidents at Pressurized Water Reactors, September 13, 2004.

23. License Amendments 217/211, Turkey Point Units 3 and 4 - Issuance of Amendments Regarding Deletion of Technical Specifications for Hydrogen Monitors and Post-Accident Containment Vent System (TAC Nos. MB0334 and MB0335), December 20, 2001.

24. NUREG-1431, Standard Technical Specifications Westinghouse Plants, Revision 3.0.

25. ASTM E-741-2000, Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution.

26. J. Pollock (Entergy – Indian Point 3) to U.S. Nuclear Regulatory Commission, NL-08-041, "Proposed Changes to the Technical Specifications Regarding Replacement of the Sodium Hydroxide Buffer with Sodium Tetraborate Decahydrate", February 28, 2008.

27. J. Boska (NRC) to Vice President, Operations (Entergy), "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment Re: Changes to Technical Specifications to Replace Sodium Hydroxide Buffer with Sodium Tetraborate (TAC No. MD8206)," June 9, 2008.

28. L-2009-062, W. Jefferson (FPL) to NRC Document Control Desk, Turkey Point Unit 4 Response to NRC Request for Additional Information Regarding the Reponses to GL 2004 -02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, TAC No. MC4726, March 19, 2009.

29. L-2009-063, W. Jefferson (FPL) to NRC Document Control Desk, Turkey Point Unit 3 Response to NRC Request for Additional Information Regarding the Reponses to GL 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors, TAC No. MC4725, March 19, 2009.

30. NUREG-0933, Resolution of Generic Safety Issues.

31. R. Uhrig (FPL) to D. Eisenhut (NRC) (L-82-276), "Post-TMI Requirements Supplement 2 to Response to Generic Letter No. 82-05," July 9, 1982.

32. S. Varga (NRC) to R. Uhrig (FPL), "NUREG-0737, Item III.D.3.4, 'Control Room Habitability' Turkey Point Plant Units 3 and 4," November 25, 1983.

33. J. Williams (FPL) to S. Varga (NRC) (L-85-164), "NUREG 0737, Item III.D.3.4 Control Room Habitability," April 17, 1985.

34. M. Chawla (NRC) to M. Balduzzi (Entergy), "Palisades Plant – Issuance of Amendment Re: Alternative Radiological Source Term (TAC No. MD3087)," September 28, 2007.

35. J. Stang (NRC) to R. Powers (Indiana Michigan Power), "Donald C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments (TAC. Nos. MB0739 and MB0740)," October 24, 2001.

LICENSE AMENDMENT REQUEST
LAR 196

ALTERNATIVE SOURCE TERM
AND CONFORMING AMENDMENT

Technical Specification Change Markups

TS Page

ix

1-3

Insert Page 1-3a

3/4 4-26

Insert Page 3/4 4-26a

3/4 4-27

3/4 4-28

3/4 4-29

3/4 6-15 (Replaced with new TS for Recirculation pH Control System

3/4 6-16 (Text deleted)

3/4 7-17

3/4 7-25

6-17

6-18a

6-19

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DEFINITIONS

FREQUENCY NOTATION

Insert A

DOSE EQUIVALENT I-131

~~1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.~~

~~$\bar{E}$-AVERAGE DISINTEGRATION ENERGY~~

Insert B

~~1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant~~.

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GAS DECAY TANK SYSTEM

1.15 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary-to-secondary leakage).

INSERT A

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

INSERT B

<u>DOSE EQUIVALENT XE-133</u>

1.13 DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-33m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

0.25 microcuries

3.4.8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to ~~1.0 microcurie~~ per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to ~~100/Ē~~ microcuries per gram ~~of gross radioactivity~~.

447.7

DOSE EQUIVALENT XE-133

APPLICABILITY: MODES 1, 2, 3, 4 ~~and 5.~~

ACTION: and

~~MODES 1, 2 and 3*:~~

~~a. With the specific activity of the reactor coolant greater than 1 microcurie per gram DOSE EQUIVALENT I 131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4 1, be in at least HOT STANDBY with average reactor coolant temperature less than 500°F within 6 hours; and~~

~~b. With the specific activity of the reactor coolant greater than 100/Ē microcurie per gram, be in at least HOT STANDBY with average reactor coolant temperature less then 500°F within 6 hours.~~

~~MODES 1, 2, 3, 4, and 5:~~

~~With the specific activity of the reactor coolant greater than 1 microcurie per gram DOSE EQUIVALENT I 131 or greater then 100/Ē microcuries per gram, perform the sampling and analysis requirements of Item 6.a) of Table 4.4 4 until the specific activity of the reactor coolant is restored to within its limits.~~

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

Add INSERT from next page

~~*With the average reactor coolant temperature greater than or equal to 500°F.~~

INSERT

a. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is less than or equal to 60 microcuries per gram once per 4 hours.

b. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 60 microcuries per gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 0.25 microcuries per gram limit. Specification 3.0.4 is not applicable.

c. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131 for greater than or equal to 48 hours during one continuous time interval, or greater than 60 microcuries per gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

d. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 447.7 microcuries per gram limit. Specification 3.0.4 is not applicable.

e. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133 for greater than or equal to 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

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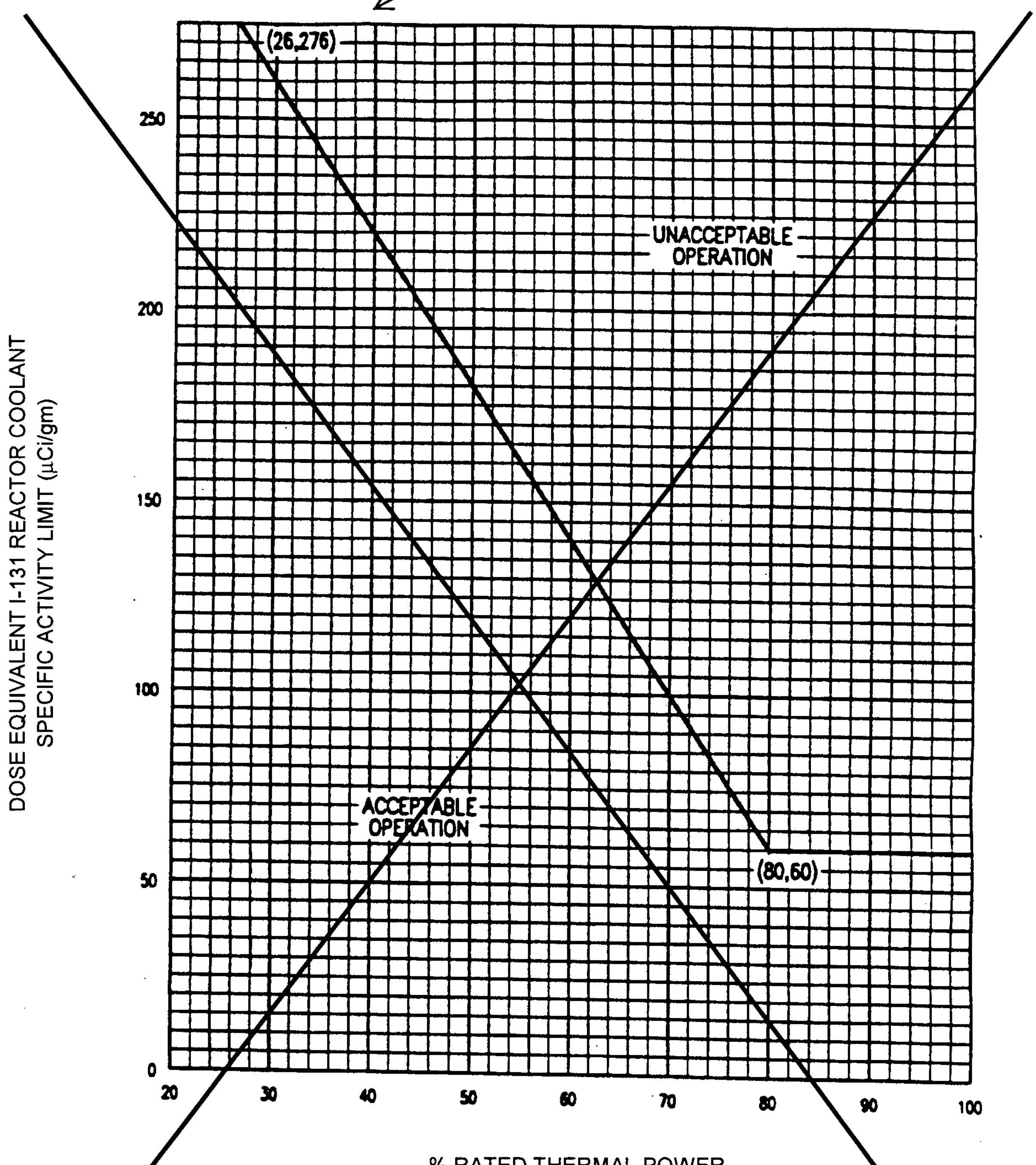

FIGURE 3.4.-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 µCi/gram DOSE EQUIVALENT I-131.

Isotopic Analysis for DOSE EQUIVALENT XE-133

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. ~~Gross Radioactivity Determination~~ NOT USED	~~At least once per 72 hours.~~	~~1, 2, 3, 4~~
2. Tritium Activity Determination	1 per 7 days.	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131 ~~Concentration~~	a) 1 per 14 days. b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period.	1
4. Radiochemical Isotopic Determination Including Gaseous Activity	Monthly	1, 2, 3, 4
5. ~~Radiochemical for $\bar{E}$ Determination~~	~~1 per 6 months*~~ 1 per 7 days	1
6. ~~Isotopic Analysis for Iodine Including I-131, I-133, and I-135~~ NOT USED	~~a) Once per 4 hours, whenever the specific activity exceeds 1 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram of gross radioactivity, and~~	~~1#, 2#, 3#, 4#, 5#~~
	~~b. One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period.~~	~~1, 2, 3~~

~~Table 4.4-4 (Continued)~~

~~TABLE NOTATIONS~~

~~* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.~~

~~# Until the specific activity of the Reactor Coolant System is restored within its limits.~~

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3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Three emergency containment filtering units shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one emergency containment filtering unit inoperable, restore the inoperable filter to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 Each emergency containment filtering unit shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release or (3) after every 720 hours of system operation by:

1) Performance of a visual inspection for foreign material and gasket deterioration, and verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP and halogenated hydrocarbons at the system flow rate of 37,500 cfm ±10%;

2) Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with applicable portions of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and performed in accordance with ASTM D3803-1989 at 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than 35% and that any charcoal failing to meet this criteria be replaced with charcoal that meets or exceeds the stated performance requirement; and

3) Verifying a system flow rate of 37,500 cfm ±10% and a pressure drop across the HEPA and charcoal filters of less than 6 inches water gauge during system operation when tested in accordance with ANSI N510-1975;

INSERT

CONTAINMENT SYSTEMS

3/4.6.2.3 RECIRCULATION pH CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 The Recirculation pH Control System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

With the Recirculation pH Control System inoperable, restore the buffering agent to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the next 72 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The Recirculation pH Control System shall be demonstrated OPERABLE:

a. At least once per 18 months by:

1. Verifying that the buffering agent baskets are in place and intact;

2. Collectively contain ≥ 11061 pounds (227 cubic feet) of sodium tetraborate decahydrate, or equivalent.

3/4.6.3 Not Used

~~SURVEILLANCE REQUIREMENTS (Continued)~~

c. ~~After maintenance affecting flow distribution, by performance of a visual inspection and an air distribution test at a system flow rate of 37,500 cfm ±10%;~~

d. ~~At least once per 18 months by:~~

1) ~~Verifying that the system starts on a Safety Injection test signal and;~~

2) ~~Verifying that the filter cooling solenoid valves can be opened by operator action and are opened automatically on a loss of flow signal.~~

e. ~~After each complete or partial replacement of a HEPA filter bank, by performance of a visual inspection for foreign material and gasket deterioration and by verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP test aerosol while operating the system at a flow rate of 37,500 cfm ±10%; and~~

f. ~~After each complete or partial replacement of a charcoal adsorber bank, by performance of a visual inspection for foreign material and gasket deterioration and by verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of halogenated hydrocarbon while operating the system at a flow rate of 37,500 cfm ±10%.~~

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% DOP and halogenated hydrocarbon removal at a system flow rate of 1000 cfm ±10%.

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ASTM D3803 - 1989 AT 30°C and 95% relative humidity, meets the methyl iodide penetration criteria of less than ~~2.5%~~ 1.25% or the charcoal be replaced with charcoal that meets or exceeds the stated performance requirement, and

3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration.

d. At least once per 12 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%;

e. At least once per 18 months by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation.

PLANT SYSTEMS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.7.9 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 Curies of noble gases ~~(considered as Xe-133 equivalent).~~

(DOSE EQUIVALENT XE-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank and the Reactor Coolant System total activity exceeds the limit of Specification 3.4.8.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. Deleted

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following deviations or exemptions:

1) Type A tests will be performed either in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972, or the guidelines of Regulatory Guide 1.163.

2) Type A testing frequency in accordance with NEI 94-01, Revision 0, Section 9.2.3, except:

   a) For Unit 3, the first Type A test performed after the November 1992 Type A test shall be performed no later than November 2007.

   b) For Unit 4, the first Type A test performed after October 1991 shall be performed no later than October 2006.

3) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment interval pressure for the design basis loss of coolant accident, P_a, is 49.9 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be ~~0.25%~~ 0.20% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding $1.0\ L_a$, the As-left leakage rate acceptance criterion is $\leq 0.75\ L_a$, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary-to-secondary accident induced leakage rate for any design basis accident, other than SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed ~~1 gpm~~ 0.60 gpm total through all SGs and ~~500 gallons per day~~ 0.20 gpm through any one SG at room temperature conditions.

3 The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall ~~be~~ plugged.

The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria:

1. For Unit 3 during Refueling Outage 23 and the subsequent operating cycles until the next scheduled inspection, and for Unit 4 during Refueling Outage 23 and the subsequent operating cycles until the next scheduled inspection, flaws found in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging.

   For Unit 3 during Refueling Outage 23 and the subsequent operating cycles until the next scheduled inspection, and for Unit 4 during Refueling Outage 23 and the subsequent operating cycles until the next scheduled inspection, all tubes with flaws identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged.

ADMINISTRATIVE CONTROLS

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

STARTUP REPORT

6.9.1 .1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.

Reports required on an annual basis shall include:

The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Fuel burnup by core region; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and (5) The time duration when the specific activity of the primary coolant exceeded ~~1.0 microcurie~~ per gram DOSE EQUIVALENT I-131.

0.25 microcuries

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

~~** This tabulation supplements the requirements of §20.2206 of 10 CFR Part 20.~~

LICENSE AMENDMENT REQUEST
LAR 196

ALTERNATIVE SOURCE TERM
AND CONFORMING AMENDMENT

Technical Specification Bases Markups
(Information Only)

TS Bases Page

10

55

57

60

61

62

67

68

Insert Page 68a

Insert Page 68b

84

89

Insert Page 89a

90

99

Insert Page 99a

101

120

L-2009-133
Attachment 2
Page 2 of 20

ENCLOSURE 1
(Page 3 of 4)

INDEX

BASES

SECTION **PAGE**

ATTACHMENT 1
(Page 44 of 111)

TECHNICAL SPECIFICATION BASES

3/4.4.5 (Cont'd)

SG tubing is subject to a variety of degradation mechanisms. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.8.4.j, Steam Generator (SG) Program, requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.j, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.4.j. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

Applicable Safety Analysis

0.20 gpm at room temperature conditions

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary-to-secondary leakage rate equal to ~~500 gpd~~ for each of the two intact SGs plus the leakage rate associated with a double-ended rupture of a single tube in the third (ruptured) SG. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere ~~via~~ safety valves ~~or~~ atmospheric dump valves. No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power.

and/or the condenser

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In the dose consequence analysis for these events the activity level in the steam discharged to the atmosphere is based on a primary-to-secondary leakage rate of ~~1~~ gpm total through all SGs and ~~500 gallons per day~~ through any one SG at ~~accident~~ conditions, or is assumed to increase to these levels as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, Reactor Coolant System Specific Activity, limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), ~~10 CFR 100 (Ref. 3),~~ 10 CFR 50.67 (Ref. 7) or the NRC approved licensing basis.

0.60

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

0.20 gpm

room temperature

initially via the condenser steam jet air ejectors (SJAE) then via the main steam

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.5 (Cont'd)

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

0.20 gpm

0.60

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analyses assume that accident leakage does not exceed ~~1~~ gpm total through all SGs and ~~500 gallons per day~~ through any one of the three SGs at ~~accident~~ conditions. The accident induced leakage rate includes any primary to secondary leakage ~~existing~~ prior to the accident in addition to primary to secondary leakage induced during the accident.

room temperature

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LOC 3.4.6.2 and limits primary-to-secondary leakage through any one SG to 150 gpd at room temperature. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Applicability

SG tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

Reactor Coolant System conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

Actions

The ACTIONS are modified by a Note clarifying that the ACTIONS may be entered independently for each SG tube. This is acceptable because the ACTIONS provide appropriate compensatory actions for each affected SG tube. Complying with the ACTIONS may allow for continued operation, and ~~subsequent~~ affected SG tubes are governed by subsequent ACTION entry and application.

any additional

TECHNICAL SPECIFICATION BASES

3/4.4.5 (Cont'd)

References

1. NEI 97-06, Steam Generator Program Guidelines
2. 10 CFR 50 Appendix A, GDC 19
3. ~~10 CRF 100~~ Not Used
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
5. Draft Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes, August 1976
6. EPRI Pressurized Water Reactor Steam Generator Examination Guidelines
7. 10 CFR 50.67, Accident ~~source term~~ Source Term

3/4.4.6 Reactor Coolant System Leakage

3/4.4.6.1 Leakage Detection Systems

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary to the containment. The containment sump level system is the normal sump level instrumentation. The Post Accident Containment Water Level Monitor - Narrow range instrumentation also functions as a sump level monitoring system. In addition, gross leakage will be detected by changes in makeup water requirements, visual inspection, and audible detection. Leakage to other systems will be detected by activity changes (e.g., within the component cooling system) or water inventory changes (e.g., tank levels).

Background

Components that contain or transport the coolant to or from the reactor core make up the Reactor Coolant System (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant Leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational Leakage LCO is to limit system operation in the presence of Leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

ATTACHMENT 1
(Page 50 of 111)

TECHNICAL SPECIFICATION BASES

3/4.4.6.2 Operational Leakage

10 CFR 50, Appendix A, GDC (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

Applicable Safety Analyses

The primary-to-secondary leakage safety analysis assumption for individual events varies. The assumption varies depending on whether the primary-to-secondary leakage from a single steam generator (SG) can adversely affect the dose consequences for the event. In which case, the affected SG is assumed to have the maximum allowable leakage (~~500 gallons per day~~ 0.20 gpm at room temperature). Collectively, however, the safety analyses for events resulting in steam discharge to the atmosphere assume that primary-to-secondary leakage from all steam generators (SGs) is ~~1~~ 0.60 gpm total and ~~500 gallons per day~~ 0.20 gpm at room temperature conditions through any one SG ~~accident conditions or increases to these levels as a result of accident conditions~~. The LCO requirement to limit primary-to-secondary leakage through any one SG to less than or equal to 150 gpd at room temperature is significantly less than the conditions assumed in the safety analysis.

For Control Room doses, primary-to-secondary ~~Primary to secondary~~ leakage is a factor in the dose releases outside containment resulting from a ~~locked rotor~~ RCCA Ejection accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a SG tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR ~~(Ref. 3)~~ analysis for SGTR assumes the contaminated secondary fluid is released to the atmosphere initially via the condenser SJAE and then via the atmospheric dump valves and/or main steam safety valves for a limited period of time. Operator action is taken to isolate the affected SG within the time period. ~~The 500 gallons per day primary to secondary leakage in each of the two intact SGs at accident conditions in the safety analysis assumption is relatively inconsequential.~~

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.4.6.2 (Cont'd)

RCCA Ejection

RCCA Ejection

Accidents for which the radiation dose release path is primary-to-secondary leakage, the ~~locked rotor~~ accident is more limiting for site radiation dose releases. The safety analysis for the ~~locked rotor~~ accident assumes that primary-to-secondary leakage from all SGs is ~~1~~ gpm total. The dose consequences resulting from the ~~locked rotor~~ accident are well within the limits defined in ~~10 CFR 100 or the NRC approved licensing basis (i.e., a small fraction of these limits)~~.

0.60

10 CFR 50.67.

RCCA Ejection

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

RCS operational leakage shall be limited to:

a. Pressure Boundary Leakage

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. Unidentified Leakage

One gallon per minute (gpm) of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Identified Leakage

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the RCS Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leak-off (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

TECHNICAL SPECIFICATION BASES

3/4.4.7 Chemistry

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

~~3/4.4.8 Specific Activity~~

~~The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state primary-to-secondary steam generator leakage rate of 500 gpd through each of the two intact steam generators. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Turkey Point site, Units 3 and 4 site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.~~

~~The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.~~

ADD INSERT
Pages 68a and 68b

ATTACHMENT 1
(Page 57 of 111)

TECHNICAL SPECIFICATION BASES

~~3/4.4.8 (Cont'd)~~

~~The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I–131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 30 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 30 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity~~.

~~Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.~~

~~Reducing Tavg to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained~~.

3/4.4.9 Pressure/Temperature Limits

All components in the RCS are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are induced by normal load transients, reactor trips and startup and shutdown operations. During RCS heatup and cooldown, the temperature and pressure changes must be limited to be consistent with design assumptions and to satisfy stress limits for brittle fracture.

3/4 4.8 Specific Activity

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67. Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB), steam generator tube rupture (SGTR), rod cluster control assembly (RCCA) ejection, or locked rotor accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate SRP acceptance criteria.

The safety analyses assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 0.20 gpm at room temperature exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.1.4, " Specific Activity."

The SLB and SGTR safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 0.25 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be 447.7 μCi/gm DOSE EQUIVALENT XE-133.

The RCS Specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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The iodine specific activity in the reactor coolant is limited to 0.25 µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 447.7 µCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that the offsite and control room doses will meet the appropriate SRP acceptance criteria.

The SLB, SGTR, locked rotor, and RCCA ejection accident analyses show that the calculated doses are within limits. Violation of the LCO may result in reactor coolant activity levels that could, in the event of any one of these accidents, lead to doses that exceed the acceptance criteria.

The ACTIONS permit limited operation when DOSE EQUIVALENT I-131 is greater than 0.25 µCi/gm and less than 60 µCi/gm. The Actions require sampling within 4 hours and every 4 hours following to establish a trend.

One surveillance requires performing a gamma isotropic analysis as a measure of noble gas specific activity of the reactor coolant at least once per 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This surveillance provides an indication of any increase in the noble gas specific activity.

A second surveillance is performed to ensure that iodine specific activity remains within the LCO limit once per 14 days during normal operation and following fast power changes when iodine spiking is more apt to occur. The frequency between 2 and 6 hours after a power change of greater than15% RATED THERMAL POWER within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation.

TECHNICAL SPECIFICATION BASES

3/4.6 Containment Systems

3/4.6.1 Primary Containment

3/4.6.1.1 Containment Integrity

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR ~~Part 100~~ 50.67 during accident conditions.

Note that some penetrations do not fall under Technical Specification 3.6.1.1. For example Penetration 38 is an electrical penetration only, closed by virtue of its seals, and therefore, nothing needs to happen to close the penetration during accident conditions; it is considered already closed. A passive failure would be required in order to get communication between the containment atmosphere and the outside atmosphere through this penetration (Turkey Point's license does not require consideration of passive failures). Similarly, closed systems inside containment already satisfy the requirement for CONTAINMENT INTEGRITY, so Tech Spec 3.6.1.1 does not apply to them at all (unless the piping itself is breached, which would be a passive failure).

The primary CONTAINMENT INTEGRITY requirement of Technical Specification 3.6.1.1 is modified by a Note allowing containment penetrations to be unisolated by opening the associated valves and airlocks under Administrative Controls when necessary to perform surveillance, testing requirements, and/or corrective maintenance. The Note also enforces compliance with Specification 3.6.4, in conjunction with Specification 3.6.1.1. The Administrative Controls shall consist of a dedicated person that is assigned the responsibility to close the valve(s) in the event of an emergency or when operation is complete.

The activity which places the pressurizer steam space vent in service to remove noncondensable gases from the Reactor Coolant System (RCS) is considered a corrective maintenance activity for the purposes of Technical Specification 3.6.1.1 compliance. Nitrogen, the primary noncondensable gas contributor, enters the RCS during refueling outages when it is used as a cover gas in the volume control tank and from exposure of the reactor coolant to atmospheric conditions during the refueling sequence. The pressurizer heaters drive these gases out of solution during RCS heatup and they collect in the pressurizer steam space. This causes two problems. First, the pressurizer does not respond to spray actuation, creating a "hard bubble." Second, the noncondensable gases migrate to the reference leg condensing pots of the pressurizer level instruments, preventing reference leg makeup and proper indication of pressurizer liquid level. The pressurizer steam space venting activity is necessary to correct these problems and achieve and maintain acceptable nitrogen levels in the RCS. Accordingly, opening the containment penetration isolation valves needed to conduct the pressurizer steam space venting maintenance activity is typically performed under Administrative Controls in accordance with the provisions of the noted exception to the CONTAINMENT INTEGRITY requirement of Technical Specification 3.6.1.1.

TECHNICAL SPECIFICATION BASES

3/4.6.2.1 (Cont'd)

When PC-600/-601 are calibrated, a test signal is supplied to each circuit to check operation of the relays and annunciators operated by subject controllers. This test signal will prevent MOVs 862A, 862B, 863A, 863B from opening. Therefore, it is appropriate to tag out the MOV breakers, and enter Technical Specification Action Statement 3.5.2.a. and 3.6.2.1 when calibrating PC-600/-601.

3/4.6.2.2 Emergency Containment Cooling System

The OPERABILITY of the Emergency Containment Cooling (ECC) System ensures that the heat removal capacity is maintained with acceptable ranges following postulated design basis accidents. To support both containment integrity safety analyses and component cooling water thermal analysis, a maximum of two ECCs can receive an automatic start signal following generation of a safety injection (SI) signal (one ECC receives an A train SI signal and another ECC receives a B train SI signal). To support post-LOCA long-term containment pressure/temperature analyses, a maximum of two ECCs are required to operate. The third (swing) ECC is required to be OPERABLE to support manual starting following a postulated LOCA event for containment pressure/temperature suppression.

The allowable out-of-service time requirements for the Containment Cooling System have been maintained consistent with that assigned other inoperable ESF equipment and do not reflect the additional redundancy in cooling capability provided by the Containment Spray System.

The surveillance requirement for ECC flow is verified by correlating the test configuration value with the design basis assumptions for system configuration and flow. An 18-month surveillance interval is acceptable based on the use of water from the CCW system, which results in a low risk of heat exchanger tube fouling.

INSERT A

~~3/4.6.3 Emergency Containment Filtering System~~

~~The OPERABILITY of the Emergency Containment Filtering System ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. System components are not subject to rapid deterioration. Visual inspection and operating/performance tests after maintenance, prolonged operation, and at the required frequencies provide assurances of system reliability and will prevent system failure. In-situ filter performance tests are conducted in accordance with the methodology and intent of ANSI N510-1975. Charcoal samples are tested using ASTM D3803-1989 in accordance with Generic Letter 99-02. The test conditions (30°C and 95% relative humidity) are as specified in the Generic Letter. Table 1 of the ASTM standard provides the tolerances that must be met during the test for each test parameter. The specified methyl iodide penetration value is based on the assumptions used in the LOCA analysis with a safety factor of 2. Technical Specification 3.6.3 requires three ECFs to be OPERABLE in Modes 1, 2, 3, and 4. Surveillance Requirement 4.6.3.d.2) states that each ECF be demonstrated OPERABLE... at least once per 18 months... by verifying that the filter cooling solenoids can be opened by operator action and are opened automatically on a loss of flow signal.~~

3/4.6.2.3 RECIRCULATION pH CONTROL SYSTEM

The Recirculation pH Control System is a passive safeguard consisting of 10 stainless steel wire mesh baskets (2 large and 8 small) containing sodium tetraborate decahydrate (NaTB) located in the containment basement (14' elevation). The initial containment spray will be boric acid solution from the refueling water storage tank. The recirculation pH control system adds NaTB to the containment sump when the level of boric acid solution from the containment spray and the coolant lost from the Reactor Coolant System rises above the bottom of the buffering agent baskets. As the sump level rises, the NaTB will begin to dissolve. The addition of NaTB from the buffering agent baskets ensures the containment sump pH will be greater than 7.0. The resultant alkaline pH of the spray enhances the ability of the recirculated spray to scavenge fission products from the containment atmosphere. The alkaline pH in the recirculation sump water minimizes the evolution of iodine and minimizes the occurrence of chloride and caustic stress corrosion on stainless steel piping systems exposed to the solution.

The OPERABILITY of the recirculation pH control system ensures that there is sufficient NaTB available in the containment to ensure a sump pH greater than 7.0 during the recirculation phase of a postulated LOCA. The baskets will not interact with surrounding equipment during a seismic event.

To achieve this, the analysis considered the minimum and maximum quantities of boron and borated water, the time-dependent post-LOCA sump temperatures, and the minimum time to reach switchover conditions. In addition, the formation of acid from radiolysis of air and water, radiolysis of chloride bearing electrical cable insulation and jacketing, and spilled reactor coolant were considered. Since, not all of the NaTB will be dissolved at the onset of containment spray recirculation, the final long-term pH was also calculated. The table below provides a summary of the quantities of NaTB and resulting pH values at the onset of recirculation and long-term.

Case No.	Purpose Min/Max pH	Small Basket Height (ft)	Large Basket Height (ft)	Initial NaTB Mass (lbm)	Dissolved Mass prior to Recirc (lbm)	pH at Recirc	Long-Term pH
1	Min	2.5	2.77	14264	4687	7.005	7.396 (min)
2	Min	1.9167	2.1867	11061	4637	7.000	7.241 (min)
3	Max	2.5	2.77	17034*	17034	8.105	8.108 (max)

*This case conservatively assumed that an additional small basket is installed in containment.

As shown, to satisfy the surveillance requirement, the two large baskets and eight small baskets must contain a combined mass of greater than 11,061 lbm of NaTB to ensure the minimum sump pH of 7.0 at the onset of spray recirculation. The large baskets have a length and width of 54 inches, and a height of 33.25 inches and elevated 3.5 inches above the containment floor. The smaller baskets have a length and width of 36 inches and a height of 30 inches and elevated 6.5 inches above the containment floor. Varying basket dimensions or elevation (e.g. basket leg height) impacts the surface to volume ratio and changes the time the NaTB is in contact with containment sump water. For instance, shorter legs would allow the NaTB to contact containment sump water sooner, therefore increasing the pH at the onset of recirculation. Longer legs, however, would reduce the pH at the onset of recirculation. The level of NaTB in the baskets required to provide an equilibrium sump solution pH greater than 7.0 is seven inches from the top of the basket; 26.25 inches for the large baskets and 23 inches for the small baskets from the bottom of the basket. The 18 month frequency for Surveillance Requirement 4.6.2.3 is sufficient to ensure that the stainless steel buffering agent baskets are intact and contain the required quantity of NaTB.

TECHNICAL SPECIFICATION BASES

~~3/4.6.3 (Cont'd)~~

~~The Technical Specification does not require that both independent trains of ECF dousing components be OPERABLE to support the ECFs. Disabling one train of ECF dousing components does not render the associated ECF inoperable.~~

~~The UFSAR states that the design requirement for the ECF system is to reduce the iodine concentration in the containment atmosphere following a MHA, to levels ensuring that the off-site dose will not exceed the guidelines of 10 CFR 100 at the site boundary. Details of the site boundary dose calculations are given in Section 14.3.5 of the UFSAR.~~

~~Following a loss of coolant accident, a safety injection signal will automatically energize motor control circuits to start the three filter unit fans. If outside power or full emergency power is available, all three filter units are started (only two are required). If electric power is limited due to the failure of an emergency diesel generator, two of the three units are started.~~

~~A borated water spray system is installed in each filter unit to dissipate the radioactive decay heat and initiated by the loss of air flow through the filter unit, such as failure of the fan. The Design Basis Document for the ECF system states that radioactive decay heat removal by dousing the ECF charcoal bed with containment spray water on ECF fan failure is a Quality Related function. As such, single failure criteria do not apply to the ECF spray system components because:~~

1) ~~Dousing is not required for the ECF to perform its safety related function of removing radioactive iodine and methyl iodide from the containment atmosphere,~~

2) ~~Dousing is not required to maintain offsite doses below 10CFR100 limits, and~~

3) ~~The ECF system can perform its safety related functions with any single failure without requiring dousing.~~

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.7.5 (Cont'd)

The Control Room Emergency Ventilation System is considered to be OPERABLE (Ref: JPN–PTN–SENP-92-017) when 1) Three air handling units (AHUs) (one of each of the three air conditioning units) are operable, 2) Two condensing units (two out of three available condensers) are operable, 3) One recirculation filter unit is operable, 4) Two recirculation fans operable, and 5) Associated dampers are operable. The reason three AHUs are required is that in the event of a single failure, only two AHUs would be available to supply air to the suction of the recirculation filter and fan. This is the configuration tested to support Technical Specification operability for flow through the emergency charcoal filter. Taking one AHU out of service renders the system incapable of operating in accordance with the tested configuration assuming an accident and a single failure (i.e., only one air handling unit available instead of the two assumed by the analysis). Any one of the three condensing (air conditioning) units is capable of maintaining the control room equipment within its environmental limits for temperature and humidity. Thus, one condensing unit can be taken out of service without impacting the ability of the Control Room Emergency Ventilation System to accomplish its intended function under single failure conditions.

Add INSERT 1 next page

System components are not subject to rapid deterioration, having lifetimes of many years, even under continuous flow conditions. Visual inspection and operating tests provide assurance of system reliability and will ensure early detection of conditions which could cause the system to fail or operate improperly. The filters performance tests prove that filters have been properly installed, that no deterioration or damage has occurred, and that all components and subsystems operate properly. The in-situ tests are performed in accordance with the methodology and intent of ANSI N510 (1975) and provide assurance that filter performance has not deteriorated below returned specification values due to aging, contamination, or other effects. Charcoal samples are tested using ASTM D3803-1989 in accordance with Generic Letter 99-02. The test conditions (30°C and 95% relative humidity) are as specified in the Generic Letter. Table 1 of the ASTM standard provides the tolerances that must be met during the test for each test parameter. ~~The specified methyl iodide penetration value is based on the assumptions used in the LOCA Analysis.~~

3/4.7.6 Snubbers

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The specified methyl iodide penetration value is based on the assumptions used in the LOCA analysis with a safety factor of 2 applied.

INSERT 1

In addition, the Control Room Emergency Ventilation System also includes the emergency air intakes, located beyond the southeast and northeast corners of the Auxiliary Building. The Control Room Emergency Ventilation System emergency air intakes are considered OPERABLE when: 1) both flow paths are available, 2) have balanced intake flow rates and 3) flow path capable of drawing outside makeup air from only the analyzed intake locations. The alternate source term radiological analyses assume both emergency air intake flow paths are available with parallel dampers ensuring outside makeup air can be drawn through both intake locations during a design basis accident and a single active failure. These analyses rely on a provision in Regulatory Guide 1.194 Section 3.3.2 that allows a reduction in the atmospheric dispersion factors (X/Qs) for dual intake arrangements with balanced flow rates to one half the more limiting X/Q value provided the two intakes are not within the same wind direction window for each release / receptor location. Accordingly, any maintenance on the emergency intake dampers or associated duct work that would prevent the Control Room Emergency Ventilation system from accomplishing these functions would require entering the action statement.

ATTACHMENT 1
(Page 90 of 111)

TECHNICAL SPECIFICATION BASES

3/4.7.8 Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the GAS DECAY TANK SYSTEM (as measured in the inservice gas decay tank) is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.7.9 Gas Decay Tanks

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each Gas Decay Tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed ~~0.5~~ 0.1 rem.

3/4.8 Electrical Power Systems

3/4.8.1, 3/4.8.2,
& 3/4.8.3 A.C. Sources, D.C. Sources, and Onsite Power Distribution

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) The safe shutdown of the facility, and (2) The mitigation and control of accident conditions within the facility.

The loss of an associated diesel generator for systems, subsystems, trains, components or devices does not result in the systems, subsystems, trains, components or devices being considered inoperable for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation for the affected unit provided (1) Its corresponding normal power source is OPERABLE; and (2) Its redundant systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generators as a source emergency power to meet all applicable LCOs are OPERABLE. This allows operation to be governed by the time limits of the ACTION statement associated with the inoperable diesel generator, not the individual ACTION statements for each system, subsystem, train, component or device. However, due to the existence of shared systems, there are certain conditions that require special provisions. These provisions are stipulated in the appropriate LCOs as needed.

TECHNICAL SPECIFICATION BASES

3/4.9.9 (Cont'd)

A normal refueling consists of 2 core alteration sequences: unloading the core, and reloading the core, typically with a suspension of core alterations in between. The core unload sequence begins with control rod unlatching, followed by removal of upper internals, followed by unloading fuel assemblies to the SFP. The core reload sequence consists of reloading fuel assemblies from the SFP, followed by upper internals installation, followed by latching control rods. Therefore, if the Containment Ventilation Isolation System is demonstrated OPERABLE at least once per 7 days following the specified testing within 100 hours prior to the start of control rod unlatching, then Containment Ventilation Isolation System operability need not be demonstrated within 100 hours prior to the start of core reload. Otherwise, the specified testing is required to be performed within 100 hours prior to the start of core reload.

3/4.9.10 &
3/4.9.11 Water Level – Reactor Vessel And Storage Pool

The restrictions on minimum water level ensure that sufficient shielding will be available during fuel movement and for removal of iodine in the event of a fuel handling accident. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 Handling Of Spent Fuel Cask

Limiting spent fuel decay time from last time critical to a minimum of 1,525 hours prior to moving a spent fuel cask into the spent fuel pit will ensure that potential offsite doses are a fraction of ~~10 CFR Part 100~~ limits should a dropped cask strike the stored fuel assemblies.

The restriction to allow only a single element cask to be moved into the spent fuel pit will ensure the maintenance of water inventory in the unlikely event of an uncontrolled cask descent. Use of a single element cask which nominally weighs about twenty-five tons will also increase crane safety margins by about a factor of four.

Requiring that spent fuel decay time from last time critical be at least 120 days prior to moving a fuel assembly outside the fuel storage pit in a shipping cask will ensure that potential offsite doses are a fraction of ~~10 CFR 100~~ limits should a dropped cask and ruptured fuel assembly release activity directly to the atmosphere.

3/4.9.13 Radiation Monitoring

The OPERABILITY of the containment radiation monitors ensures continuous monitoring of radiation levels to provide immediate indication of an unsafe condition.

10 CFR 50.67

10 CFR 50.67

LICENSE AMENDMENT REQUEST

**ALTERNATIVE SOURCE TERM
AND CONFORMING AMENDMENT
LAR 196**

ATTACHMENT 3

NUMERICAL APPLICATIONS INC.

NAI REPORT RELEASE

NAI REPORT NO. NAI-1396-045

**AST LICENSING TECHNICAL REPORT
FOR TURKEY POINT UNITS 3 AND 4
REVISION 1**

DATED JUNE 18, 2009

NAI Report Release

Report Number: NAI-1396-045

Revision Number: 1

Title: AST Licensing Technical Report for Turkey Point Units 3 and 4

Description:

> This report documents the results of the analyses and evaluations performed by Numerical Applications, Inc. in support of the Turkey Point licensing project to implement Alternative Source Term (AST). Design basis accidents and radiological consequences are evaluated using the AST methodology to support control room habitability in the event of increases in unfiltered inleakage. The analyses and evaluations performed by NAI are based on the guidance of Regulatory Guide 1.183.

Mark A Pope 6/18/09
Author (Mark Pope) Date

Jim Harrell 6/18/09
Reviewer (Jim Harrell) Date

TGeorge Jun 18, 2009
NAI Management (Tom George) Date

Table of Contents

Figures and Tables

1.0 Radiological Consequences Utilizing the Alternative Source Term Methodology

1.1 Introduction

The current Turkey Point licensing basis for radiological consequences analyses of accidents discussed in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) is based on methodologies and assumptions that are primarily derived from Technical Information Document (TID)-14844 and other early guidance.

Regulatory Guide 1.183 provides guidance on application of an Alternative Source Term (AST) in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10CFR50.67. Because of advances made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents, 10CFR50.67 is issued to allow holders of operating licenses to voluntarily revise the traditional accident source term used in the design basis accident (DBA) radiological consequence analyses with an alternative source term.

1.2 Evaluation Overview and Objective

As documented in NEI 99-03 and Generic Letter 2003-01, several nuclear plants performed testing on control room unfiltered air inleakage that demonstrated leakage rates in excess of amounts assumed in the current accident analyses. The AST methodology as established in Reg. Guide 1.183 is being used to calculate the offsite and control room radiological consequences for Turkey Point Units 3 and 4 to support the control room habitability program by addressing the radiological impact of potential increases in control room unfiltered air inleakage. In all cases, the most limiting configuration between Units 3 and 4 is utilized for the radiological analyses.

The following limiting UFSAR Chapter 14 accidents are analyzed:

- Loss-of-Coolant Accident (LOCA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Locked Rotor
- Rod Cluster Control Assembly (RCCA) Ejection
- Fuel Handling Accident (FHA)
- Waste Gas Decay Tank (WGDT) Rupture
- Spent Fuel Cask Drop Accident

Note that although RG 1.183 does not include the WGDT or Spent Fuel Cask Drop accidents, these events were included in the AST analysis to incorporate new atmospheric dispersion factors and to evaluate the dose consequences using the TEDE criteria consistent with the other limiting UFSAR events.

Each accident and the specific input and assumptions are described in Section 2.0 of this report. These analyses provide for a bounding allowable control room unfiltered air inleakage of 115 cfm. The use of 115 cfm as a design basis value was established to be above the unfiltered inleakage value determined through testing or analysis consistent with the resolution of issues identified in NEI 99-03 and Generic Letter 2003-01. The control room inleakage testing performed at Turkey Point in 2003 has resulted in less than 10 cfm of unfiltered inleakage when all uncertainties and tolerances were considered. Therefore, there is significant margin between the bounding dose analysis inleakage value of 115 cfm and the measured control room unfiltered inleakage.

1.3 Proposed Changes to the Turkey Point Licensing Basis

Florida Power and Light (FP&L) Company proposes to revise the Turkey Point licensing basis to implement the AST, described in Reg. Guide 1.183, through reanalysis of the radiological consequences of the UFSAR Chapter 14 accidents listed in Section 1.2 above. This is a full implementation of the AST as described in Section 1.1.3 of the Reg. Guide. Key elements of the revised analysis include:

- The total effective dose equivalent (TEDE) acceptance criterion of 10CFR50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10CFR100.11.
- New onsite (Control Room) and offsite atmospheric dispersion factors are developed.
- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12, respectively.
- A bounding value for control room unfiltered air inleakage was established by increasing the inleakage until the dose acceptance criteria for the limiting event (LOCA) was approached.
- A primary coolant specific activity that is more restrictive than the current Technical Specification limit is utilized.
- A steam generator tube leakage rate is used that is more restrictive than the current Technical Specification program limit for primary-to-secondary accident induced leakage
- A containment leakage value that is more restrictive than the current Technical Specification limit is utilized.
- Sump pH control is provided by sodium tetraborate decahydrate (NaTB) baskets.
- Containment ESF filter units are not being credited.
- The Containment Spray System and Recirculation pH Sump Control System are being credited for post-LOCA iodine removal.
- For the Secondary RCCA Ejection, FHA with a Fuel Building release, and Spent Fuel Cask Drop events, credit is taken for the operators to manually isolate the Control Room Ventilation System within 30 minutes.
- The Control Room emergency intakes are being relocated beyond the southeast and northeast and corners of the Reactor Auxiliary Building.
- The Control Room Emergency Ventilation System methyl iodine penetration criterion is being reduced from 2.5% to 1.25%.

Accordingly, the following changes to the Turkey Point Units 3 and 4 Technical Specifications (TS) are proposed:

- The definition of Dose Equivalent I-131 in Section 1.12 is revised to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989, as the source of effective dose conversion factors.

- The definition of Average Disintegration Energy in Section 1.13 is replaced with the definition of Dose Equivalent Xe-133 based upon effective dose conversion factors from Table III.1 of Federal Guidance Report No. 12 (FGR 12), "External Exposure to Radionuclides in Air, Water, and Soil," 1983.

- The Reactor Coolant System (RCS) specific activity limit for dose equivalent Iodine-131 (DE I-131), stated in Limiting Condition for Operation (LCO) 3.4.8.a, is reduced from 1 microcurie per gram to 0.25 microcurie per gram.

- The Reactor Coolant System specific activity limit for gross radioactivity stated in Limiting Condition for Operation (LCO) 3.4.8.b, is changed from 100/E-bar to dose equivalent Xenon-133. The DE Xe-133 limit of 447.7 microcuries per gram is established at a value which maintains the current maximum allowable RCS activity level based upon E-bar.

- The methyl iodine penetration criteria for the Control Room Emergency Ventilation System charcoal adsorber in surveillance requirement 4.7.5.c.2 is reduced from 2.5% to 1.25%.

- The terminology used in Limiting Condition for Operation (LCO) 3.7.9 to set the maximum contents of the Waste Gas Decay Tank is clarified to read 'DOSE EQUIVALENT Xe-133.'

- The maximum allowable containment leakage rate acceptance criterion stated in TS 6.8.4.h, "Containment Leakage Rate Testing Program," is reduced from 0.25% to 0.20% of containment air weight per day.

- The primary-to-secondary steam generator leakage rate assumed in the accident analysis and specified in the accident induced leakage performance criterion of TS 6.8.4.j.b.2, "Steam Generator Program," is reduced from "1 gpm through all steam generators and 500 gpd through any one steam generator" [at accident conditions] to "0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions."

- A method for controlling the pH of the post-LOCA containment sump solution using sodium tetraborate decahydrate is being proposed. This passive system will consist of baskets of NaTB in the lower regions of the containment. Appropriate technical specifications and surveillance requirements are proposed for Section 3/4.6 "Containment Systems."

- Operability requirements for emergency containment filter units in Section 3/4.6.3 will be deleted.

1.4 <u>Compliance with Regulatory Guidelines</u>

The revised Turkey Point accident analyses addressed in this report follow the guidance provided in Reg. Guide 1.183. Assumptions and methods utilized in this analysis for which no specific guidance is provided in Reg. Guide 1.183, but for which a regulatory precedent has been established, are as follows:

- Selection of the WGDT Rupture offsite dose consequences acceptance criteria for the EAB is based on the Branch Technical Position 11-5 of the Standard Review Plan, which gives the total body exposure acceptance criteria for an individual at the nearest exclusion area boundary following a waste gas system failure as 0.1 rem. Additional guidance related to the application of the AST methodology to the analysis of this event is given in Issue #11 of NRC Regulatory Issue Summary 2006-04. RIS 2006-04 sets the AST dose limit to a member of the public for this event at 100 mrem TEDE. Since RIS 2006-04 associates this acceptance criteria with the annual limit established in 10CFR20, the EAB dose will be evaluated over the duration of the event rather than for the worst two-hour period. Branch Technical Position 11-5 does not require the dose consequences to be evaluated at the LPZ or the control room. These locations will be evaluated for completeness. The LPZ dose will be evaluated against the EAB acceptance criteria of 0.1 rem TEDE, and the TEDE dose limit of 5 rem from 10CFR50.67 will be applied to the control room consistent with the other AST events.

- Regulatory Guide 1.183 does not provide specific guidance for the Spent Fuel Cask Drop event. However, the guidance for the Fuel Handling Accident in Appendix B of the Reg. Guide is judged to be closely applicable to the conditions of this event. Therefore, the evaluation of this event will conform to the positions of Reg. Guide 1.183, Appendix B, and the acceptance criteria for the Fuel Handling Event from Section 4.4 and Table 6 will be applied.

- Use of the MicroShield code to develop direct shine doses to the Control Room. MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis. It is qualified for this application and has been used to support licensing submittals that have been accepted by the NRC. Precedent for this use of MicroShield is established in the Duane Arnold Energy Center submittal dated October 19, 2000 and associated NRC Safety Evaluation dated July 31, 2001.

1.5 Computer Codes

The following computer codes are used in performing the Alternative Source Term analyses:

Computer Code	Version	Reference	Purpose
ARCON96	June 1997	5.12	Atmospheric Dispersion Factors
MicroShield	5.05	5.13	Direct Shine Dose Calculations
ORIGEN	2.1	5.14	Core Fission Product Inventory
PAVAN	2.0	5.15	Atmospheric Dispersion Factors
GOTHIC	7.1	5.16 - 5.18	Containment Mixing
RADTRAD-NAI	1.0p3	5.19	Radiological Dose Calculations

1.5.1 ARCON96 – used to calculate relative concentrations (*X/Q* factors) in plumes from nuclear power plants at control room intakes in the vicinity of the release point using plant meteorological data.

1.5.2 MicroShield – used to analyze shielding and estimate exposure from gamma radiation.

1.5.3 ORIGEN – used for calculating the buildup, decay, and processing of radioactive materials.

1.5.4 PAVAN – provides relative air concentration (*X/Q*) values as functions of direction for various time periods at the EAB and LPZ boundaries assuming ground-level releases or elevated releases from freestanding stacks.

1.5.5 GOTHIC – The GOTHIC containment analysis code is used to determine the containment mixing due to actuation of the containment sprays and operation of the Emergency Containment Coolers.

1.5.6 RADTRAD-NAI – estimates the radiological doses at offsite locations and in the control room of nuclear power plants as consequences of postulated accidents. The code considers the timing, physical form (i.e., vapor or aerosol) and chemical species of the radioactive material released into the environment.

RADTRAD-NAI began with versions 3.01 and 3.02 of the NRC's RADTRAD computer code, originally developed by Sandia National Laboratory (SNL). The code is was modified to compile on a UNIX system. Once compiled, an extensive design review/verification and validation process was completed on the code and documentation. The subject of the review also included the source code for the solver, which was made available in a separate distribution from the NRC. RADTRAD-NAI validation was performed with three different types of tests:

- Comparison of selected Acceptance Test Case results with Excel spreadsheet solutions and hand solutions,
- Separate effects tests, and
- Industry examples.
- The industry examples included prior AST submittals by BWRs and PWRs, as well as other plant examples.

In addition to reviewing the code and incorporating error corrections, several software revisions were made. One revision involved the consideration of noble gases generated by decay of isotopes on filters that are returned to the downstream compartment. Another revision involved the modification of the dose conversion and nuclide inventory files to account for 107 isotopes to assure that significant dose contributors were addressed. The dose conversion factors used by RADTRAD-NAI are from Federal Guidance Report Nos. 11 and 12 (FGR 11and FGR 12).

RADTRAD-NAI was developed and is maintained under Numerical Applications' 10CFR50 Appendix B program.

1.6 Radiological Evaluation Methodology

1.6.1 Analysis Input Assumptions

Common analysis input assumptions include those for the control room ventilation system and dose calculation model (Section 1.6.3), direct shine dose (Section 1.6.5), radiation source terms (Section 1.7), and atmospheric dispersion factors (Section 1.8). Event-specific assumptions are discussed in the event analyses in Section 2.0.

1.6.2 Acceptance Criteria

Offsite and Control Room doses must meet the guidelines of Reg. Guide 1.183 and requirements of 10CFR50.67. The acceptance criteria for specific postulated accidents are provided in Table 6 of Reg. Guide 1.183. For Waste Gas Decay Tank Rupture and Spent Fuel Cask Drop events, which are not addressed in Reg. Guide 1.183, the basis used to establish the acceptance criteria for the radiological consequences is provided in Section 1.4.

1.6.3 Control Room Ventilation System Description

The Control Room Ventilation System is required to assure control room habitability. The design of the control room envelope and overall description of the Control Room Ventilation System are discussed in the Turkey Point UFSAR, Section 9.9.1.

The Control Room Ventilation System consists of three 100% capacity air handling units and a ducted air intake and air distribution system. Outside air is drawn into the air handling units through roughing filters and cooled as required. Conditioned air is then directed back to the rooms through a supply air duct system.

Under emergency conditions, the Control Room Ventilation System has the capability to go into the recirculation mode. In the recirculation mode, fresh and recirculated air is processed through high efficiency particulate (HEPA) filters and charcoal filters to maintain the control room environment at acceptable conditions. The recirculation mode is automatically entered on receipt of an outside normal air intake high radiation signal, a containment high radiation signal or a safety injection signal.

Redundant isolation dampers at the normal outside air intake and exhaust paths are automatically closed, so that the control room envelope is isolated except for filtered fresh air makeup. The system is designed to perform its safety functions and maintain a habitable environment in the control room envelope during isolation.

The control room envelope is slightly pressurized relative to the surroundings during normal plant operation with outside air continuously introduced to the control room envelope at a rate of 1000 cfm. In the recirculation mode, the control room is pressurized at the rate of 525 cfm to maintain a positive pressure differential. Makeup air for pressurization is filtered before entering the control room. The recirculated air flow rate of 375 cfm is filtered by the same filters as the makeup air.

The net volume of the control room envelope serviced by the Control Room Ventilation System is 47,786 cubic ft.

1.6.3.1 Control Room Dose Calculation Model

The control room model includes a recirculation filter model along with filtered air intake, unfiltered air inleakage and an exhaust path. System performance, sequence, and timing of operational evolutions associated with the control room ventilation system are discussed below. Control room ventilation system parameters assumed in the analyses are provided in Table 1.6.3-1. The dispersion factors for use in modeling the control room during each mode of operation are provided in Tables 1.8.1-2 and 1.8.1-3. Control room occupancy factors and assumed breathing rates are those prescribed in Reg. Guide 1.183. Figure 1.8.1-1 provides a site sketch showing the Turkey Point plant layout, including the location of onsite potential radiological release points with respect to the control room air intakes. The elevations of release points and intakes used in the control room AST dose assessments are provided in Table 1.8.1-1.

The control room ventilation system contains a filtration system for removal of radioactive iodine and particulate material that may enter the control room during the course of the event. Calculation of the dose to operators in the control room requires modeling of various system configurations and operating evolutions of the control room ventilation system during the course of the accident. While in the short duration normal mode prior to control room normal air intake and exhaust isolation, a single inlet to the control room (on the southwest corner of the Control Building) with an unfiltered flow rate of 1000 cfm is modeled. When in the emergency/recirculation mode, the control room model will define two concurrent air inlet paths representing the defined control room ventilation system air intake and the unfiltered inleakage into the control room. In the emergency/recirculation mode, outside air enters the control room through the filtration/ventilation system from both intakes which are located near ground level off of the southeast and northeast corners of the Auxiliary Building. Based on the release point, the dispersion factors for the most conservative emergency intake are modeled. Since the control room envelope remains pressurized while in the emergency/recirculation mode, unfiltered inleakage is modeled as makeup air which bypasses the filtration system. Modeling of the control room conservatively addresses these factors as they apply to the various release locations for each analyzed event. Details of the control room modeling for each event are described in subsequent event analyses sections.

For all events, delays in switching to the emergency/recirculation mode from the normal mode are conservatively considered with respect to the time required for signal processing, relay actuation, time required for the dampers to move and the system to re-align and diesel generator start time.

1.6.4 Direct Shine Dose

The total control room dose also requires the calculation of direct shine dose contributions from control room filters, from the radioactive plume in the environment, and from the containment building. The contribution to the total dose to the operators from direct radiation sources were calculated for the LBLOCA event. The LOCA shine dose contribution is assumed to be bounding for all other events.

Direct shine dose is determined from four different sources to the control room operator after a postulated LOCA event. These sources are the containment building, the control room recirculation filters, the external cloud that envelops the control room, and from the containment purge duct penetration. All other sources of direct shine dose are considered negligible. The MicroShield 5 code is used to determine direct shine exposure to a dose point located in the control room. Each source required a different MicroShield case structure including different geometries, sources, and materials. The external cloud is assumed to have a length of 1000 meters in the MicroShield cases to approximate an infinite cloud. A series of cases is run with each structure to determine an exposure rate from the radiological source at given points in time. These sources were taken from RADTRAD-NAI runs which output the nuclide activity at selected points in time for the event. The RADTRAD-NAI output provides the time dependent results of the radioactivity retained in the control room filter components, as well as the activity inventory in the environment and the containment. A bounding control room filter inventory is established by increasing the unfiltered inleakage to produce a control room dose slightly in excess of the 5 rem TEDE dose limit and by maximizing the filter efficiency.

The RADTRAD-NAI sources were then input into the MicroShield case file where they are either used as is, or 'decayed' (once the release has stopped) in MicroShield to yield the source activity at a later point in time. The exposure results from the series of cases for each source term were then corrected for occupancy using the occupancy factors specified in Reg. Guide 1.183. The cumulative exposure and dose are subsequently calculated to yield the total 30-day direct shine dose from each source. The results of the Direct Shine Dose evaluation are presented in Table 1.6.3-2.

The applicable components of the LOCA shine dose are used as a conservative assessment of the direct shine dose contribution for all other accidents.

1.7 Radiation Source Terms

1.7.1 Fission Product Inventory

The source term data to be used in performing alternative source term (AST) analyses for Turkey Point are summarized in the following tables:

Table 1.7.2-1 - Primary Coolant Source Term
Table 1.7.3-1 - Secondary Side Source Term (non-LOCA)
Table 1.7.4-1 - Core Source Term
Table 1.7.5-1 - Fuel Handling Accident Source Term

Note that the source terms provided in the referenced tables do not include any decay before the start of the events. Decay time assumptions are applied in the RADTRAD cases for individual event analysis. For example, the RADTRAD case for the Fuel Handling Accident analysis would account for the required decay time before the movement of fuel is allowed (as determined by Technical Specifications).

The Turkey Point reactor core consists of 157 fuel assemblies. The full core isotopic inventory is determined in accordance with Reg. Guide 1.183, Regulatory Position 3.1, using the ORIGEN-2.1 isotope generation and depletion computer code (part of the SCALE-4.3 system of codes) to develop the isotopics for the specified burnup, enrichment, and burnup rates (power levels). The plant-specific isotopic source terms are developed using a bounding approach.

The assembly source term is based on 2644 MW_{th}, with 0.3% calorimetric uncertainty, or 2652 MW_{th}, which exceeds the current rated core thermal power of 2300 MW_{th}. For non-LOCA events with fuel failures, a bounding radial peaking factor of 1.65 is then applied to conservatively simulate the effect of power level differences across the core that might affect the localized fuel failures for assemblies containing the peak fission product inventory.

The following assumptions are applied to the source term calculations:

1. A conservative maximum fuel assembly uranium loading (463 kilograms) is assumed to apply to all 157 fuel assemblies in the core.

2. Sensitivity studies were performed to assess the bounding fuel enrichment and bounding burnup values.

3. Radioactive decay of fission products during refueling outages is ignored in the source term calculation.

4. When adjusting the primary coolant isotopic concentrations to achieve Technical Specification limits, the relative concentrations of fission products in the primary coolant system are assumed to remain constant.

1.7.2 Primary Coolant Source Term

The primary coolant source term for Turkey Point is calculated based upon maximum equilibrium concentrations from operation at 2652 MW_{th} with small defects in 1 percent of the fuel rod cladding. The equilibrium iodine activities were then adjusted to achieve the proposed Technical Specification 3.4.8 limit of 0.25 μCi/gm dose equivalent I-131 using the proposed Technical Specification definition of Dose Equivalent I-131 (DE I-131).

The non-iodine activities were determined by first developing a list of isotopes which satisfied the radionuclide requirements specified in Table 5 of Reg. Guide 1.183. Iodine nuclides and isotopes with half lives less than 30 days were deleted from this list in accordance with the current Technical Specification definition of E-bar. Equilibrium RCS activities based upon 1% fuel defects were combined with corrosion product activities from ANSI/ANS-18.1-1999 for Cr-51, Fe-55, F-59, and Mn-54 to calculate a total RCS specific activity. This value was found to be slightly greater than the Tech. Spec. limit of 100/E-bar. The activity for each isotope was then adjusted by a constant factor such that the sum of the adjusted activities was equal to the 100/E-bar limit. The adjusted primary coolant source term activities for the applicable Table 5 list of isotopes is presented in Table 1.7.2-1, "Primary Coolant Source Term."

A value for DE Xe-133 was calculated using the proposed Technical Specification definition and the equilibrium noble gas activities based upon 1% fuel defects. This value was corrected using the same adjustment factor needed to achieve a total specific activity equal to 100/E-bar described above. The resulting adjusted DE Xe-133 is equal to 447.7 μCi/gm. An RCS DE Xe-133 limit of 447.7 μCi/gm ensures that the maximum RCS activity will remain below the current Tech. Spec. limit based upon E-bar.

1.7.3 Secondary Side Coolant Source Term

Secondary coolant system activity is limited to a value of ≤ 0.10 μCi/gm dose equivalent I-131 in accordance with TS 3.7.1.4. Noble gases entering the secondary coolant system are assumed to be immediately released; thus the noble gas activity concentration in the secondary coolant system is assumed to be 0.0 μCi/gm. Thus, the secondary side iodine activity is 40% of the activity given in Table 1.7.2-1.

The secondary side source term is presented in Table 1.7.3-1, "Secondary Side Source Term (non-LOCA)."

1.7.4 Core Source Term

Per Section 3.1 of Reg. Guide 1.183, the inventory of fission products in the Turkey Point reactor core and available for release to the containment is based on the maximum full power operation of the core and the current licensed values for fuel enrichment and fuel burnup. The period of irradiation is selected to be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. In addition, all fuel assemblies in the core are assumed to be affected and the core average inventory is used.

The core source term is based on an "average" assembly with a bounding core average burnup of 45,000 MWD/MTU and a bounding average assembly power* of 16.892 MW_{th}. The minimum fuel enrichment is based on an historical minimum of 3.0 w/o and a bounding maximum assumed fuel enrichment of 5.0 w/o. The limiting isotopic concentration from either enrichment was used. It is conservatively assumed that a maximum assembly uranium mass of 463,000 gm applies to all of the fuel assemblies.

*Average assembly power = (2652 MW_{th})(1 / 157 assemblies) = 16.892 MW_{th} / assembly

The ORIGEN runs used cross section libraries that correspond to PWR extended burnup fuel. Decay time between cycles is conservatively ignored. For each nuclide, the bounding activity for the allowable range of enrichments is determined.

The core source term is presented in Table 1.7.4-1, "Core Source Term."

1.7.5 Fuel Handling Accident Source Term

The fuel handling accident for Turkey Point assumes the failure of one assembly; therefore, the fuel handling accident source term is based on a single "bounding" fuel assembly.

Per Section 3.1 of Reg. Guide 1.183, the source term methodology for the Fuel Handling Accident is similar to that used for developing the LOCA source term, except that for DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, a radial peaking factor of 1.65 is applied in determining the inventory of the damaged rods.

The LOCA source term is based on the activity of 157 fuel assemblies and the radial peaking factor is 1.65. Thus, based on the methodology specified in Reg. Guide 1.183, the fuel handling accident source term is derived by applying a factor of 1.65/157 to the LOCA source term.

The FHA source term is presented in Table 1.7.5-1, "Fuel Handling Accident Source Term."

1.7.6 Gap Release Fractions

The core inventory release fractions for the gap release and early in-vessel damage phases used for the design basis LOCA are provided in Reg. Guide 1.183, Regulatory Position 3.2, Table 2, "PWR Core Inventory Fraction Released into Containment." For the RCCA Ejection accident, the fraction of the core inventory assumed to be in the fuel rod gap are those event-specific source term requirements listed in Appendix H of Reg. Guide 1.183. The fraction of the core inventory assumed to be in the fuel rod gap for the Fuel Handing and Spent Fuel Cask Drop accidents are discussed below. For the other non-LOCA events, the fraction of the core inventory assumed to be in the gap are consistent with Reg. Guide 1.183, Regulatory Position 3.2, Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap."

For the Fuel Handling and Spent Fuel Cask Drop events, the gap fractions specified in Table 3 of the Reg. guide are modified to account for high burnup fuel. Footnote 11 on Table 3 establishes burnup limits for the applicability of the gap inventory for non-LOCA events. Consideration is given to fuel with a current burnup greater than 54,000 MWD/MTU which may have exceeded the 6.3 kw/ft linear heat generation rate during a previous operating cycle. This is done using the guidance of NUREG/CR-5009, which endorses the gap release fractions for fuel handling events outlined in Reg. Guide 1.25 with some modification for higher burnups. The following table lists the modified noble gas and iodine gap fractions from NUREG/CR-5009, which are approximately twice those of Reg. Guide 1.183:

Group	Fraction
I-131	0.12
Kr-85	0.30
Other Noble Gases	0.10
Other Halogens	0.10

Although only a few rods may have exceeded the burnup limits of Table 3 of Reg. Guide 1.183, these values are conservatively applied to the entire fuel assembly. This methodology was approved for use with high burnup rods in the fuel handling accident at Indian Point 3 in License Amendment No. 215.

1.8 <u>Atmospheric Dispersion (*X/Q*) Factors</u>

1.8.1 Onsite *X/Q* Determination

New *X/Q* factors for onsite release-receptor combinations are developed using the ARCON96 computer code ("Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997, RSICC Computer Code Collection No. CCC-664). Additionally, NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003, has been implemented. Reg. Guide 1.194 contains new guidance that supersedes the NUREG/CR-6331 recommendations for using certain default parameters as input. Therefore, the following changes from the default values are made:

- For surface roughness length, m, a value of 0.2 is used in lieu of the default value of 0.1, and
- For averaging sector width constant, a value of 4.3 is used in lieu of the default value of 4.0.

- A number of various release-receptor combinations are considered for the onsite control room atmospheric dispersion factors. These different cases are considered to determine the limiting release-receptor combination for the events.

Figure 1.8.1-1 provides a sketch of the general layout of Turkey Point that has been annotated to highlight the release and receptor point locations described above, among others. All releases are taken as ground releases per guidance provided in Reg. Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, February 1983.

Table 1.8.1-1, "Release-Receptor Combination Parameters for Analysis Events," provides information related to the relative elevations of the release-receptor combinations, the straight-line horizontal distance between the release point and the receptor location, and the direction (azimuth) from the receptor location to the release point. Angles are calculated based on trigonometric layout of release and receptor points in relation to the North-South and East-West axes. Plant North is aligned with True North.

Table 1.8.1-2, "Onsite Atmospheric Dispersion Factors (*X/Q*) for Analysis Events," provides the Control Room *X/Q* factors for the release-receptor combinations listed above. These factors are not corrected for occupancy. This table summarizes the *X/Q* factors for the control room intakes used in the various accident scenarios for onsite control room dose consequence analyses. Values are presented for the normal intake prior to control room isolation and for the unfavorable emergency intake during control room isolation.

Table 1.8.1-3, "Release-Receptor Point Pairs Assumed for Analysis Events," identifies the Release-Receptor pair and associated Control Room *X/Q* factors from Table 1.8.1-2 that are used in the event analyses during each of the modes of control room ventilation.

A building wake term is only applied to releases directly from the containment surface. The building area used for this wake term is 1,254 m^2. This value is calculated to be conservatively small in that the height used in the area calculation is from the highest roof elevation of a nearby building to the elevation of the bottom of the containment dome.

Section 3.3.2.2 of Reg. Guide 1.194 allows for the use of an effective X/Q for dual intake arrangements if the two intakes are not located in the same wind direction. This credit allows for a reduction in the X/Qs to the more limiting intake in proportion to the relative flow rate through the intakes. The control room emergency intakes are being relocated into separate wind sectors for all release points and will be balanced to have equal flow rates. Thus, the dual intake dilution credit enables the X/Qs to the most limiting emergency intake to be reduced by a factor of two in the event analyses.

1.8.2 Offsite *X/Q* Determination

For offsite receptor locations, the new atmospheric dispersion (*X/Q*) factors are developed using the PAVAN computer code ("PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Material from Nuclear Power Stations," NUREG/CR-2858, November 1982, RSICC Computer Code Collection No. CCC-445). The offsite maximum *X/Q* factors for the EAB and LPZ are presented in Table 1.8.2-1, "Offsite Atmospheric Dispersion Factors (*X/Q*)." In accordance with Regulatory Position 4 from NUREG/CR-2858, the maximum value from all downwind sectors for each time period are compared with the 5% overall site *X/Q* values for those boundaries, and the larger of the values are used in evaluations. Note that the 0-2 hour EAB atmospheric dispersion factor is applied to all time periods in the analyses.

All of the releases are considered ground level releases because the highest possible release elevation is 200 feet (from the plant stack). From Section 1.3.2 of Reg. Guide 1.145, a release is only considered a stack release if the release point is at a level higher than two and one-half times the height of adjacent solid structures. For the Turkey Point plant, the elevation of the top of the containment structures is given as 186 ft and 4-3/8 in. The highest possible release point is not 2.5 times higher than the adjacent containment building; therefore, all releases are considered ground level releases. As such, the release height is set equal to 10.0 meters as required by Table 3.1 of NUREG/CR-2858. The building area used for the building wake term is the same as for some of the ARCON96 onsite *X/Q* cases. The building height entered into PAVAN is the top elevation of the cylindrical portion of the containment building of 170.28 ft less the plant grade elevation of 18 ft.

1.8.3 Meteorological Data

Meteorological data over a five-year period (2003 through 2007) is used in the development of the new onsite *X/Q* factors used in the analysis. The meteorological data is converted from the raw format into the proper formatting required to create the meteorological data files for the ARCON96 runs. Five years worth of meteorological data is used which meets the guidance set forth in Section 3.1 of Reg. Guide 1.194. The raw data for 2003 through 2007 was provided in electronic format. The data from these files was manipulated within a spreadsheet for appropriate formatting for use with ARCON96.

ARCON96 analyzes the meteorological data file used and lists the total number of hours of data processed and the number of hours of missing data in the case output. A meteorological data recovery rate may be determined from this information. For the 2003 to 2007 data base, the meteorological data recovery rate is 97.5%. No regulatory guidance is provided in Reg. Guide 1.194 and NUREG/CR-6331 on the valid meteorological data recovery rate required for use in determining onsite *X/Q* values. However, Regulatory Position C.5 of Reg. Guide 1.23 requires a 90% data recovery threshold for measuring and capturing meteorological data. Clearly, the 97.5% valid meteorological data rate for the cases in this analysis exceeds the 90% data recovery limit set forth by Reg. Guide 1.23. With a data recovery rate of 97.5% and a total of five years worth of data, the contents of the meteorological data file are representative of the long-term meteorological trends at the Turkey Point site.

The meteorological data were also provided in annual joint frequency distribution format for 2005 through 2007 for the offsite analysis. The joint frequency distribution file requires the annual meteorological data to be sorted into several classifications. This is accomplished by using three classifications that include wind direction, wind speed, and atmospheric stability class. The format for the file conforms to the format provided in Table 1 of Reg. Guide 1.23, with the exceptions of a category for the wind direction and that the wind directions are listed from NNE to N instead of N to NNW. These data are provided for each year in terms of the percent of hours of that year that fell into each classification category. The data for each category (i.e. wind speed, wind direction, and stability class unique combination) were converted from percent to number of hours.

The number of hours for each classification is then rounded to the nearest whole hour. The total values for each stability class are then transposed so that the rows correspond to the wind speed bins and the columns correspond to the wind directions. The wind directions are then ordered properly so that the first column corresponds to the north wind direction and the last column corresponds to the NNW direction as required by the PAVAN code. The final ordered numbers are used in the input file for PAVAN.

The tower height at which the wind speeds are measured is 11.58 meters above plant grade. There were 81 calm hours in the three year joint frequency data. This low number of calm hours is likely due to the positioning of the Turkey Point plant and its proximity to the Atlantic Ocean. The highest windspeed category is classified in both Reg. Guide 1.23, "Onsite Meteorological Programs," February 1972 and in the raw data as 'greater than 24 mph', however the PAVAN code requires that the maximum speed for each category be input. Therefore, a 65-mph value is chosen as the upper limit on the fastest windspeed category because the raw meteorological data showed that there were no hours with windspeeds faster than 65 mph for the period under consideration.

An additional process was performed using the available meteorological to determine the average air temperature swing over a 24-hour period. This was done by combining the yearly ARCON96 data files from 1997 to 2001 and calculating the average temperature swing over any 24-hour period and the median 24-hour temperature change. The average air temperature range over the five years of meteorological data was calculated to be 10.1 °F, with a median temperature swing of 9.5 °F. The higher value is used to support determining the leakage rate from the RWST.

Another process was performed on the meteorological data used for the ARCON96 runs to determine the 95th percentile wind speed at the limiting MSSV release height. The limiting release height is the one at which the calculated 95th percentile wind speed is the greatest. To determine the 95th percentile wind speed at the MSSV release height, the meteorological data used for the ARCON96 runs was evaluated. Hourly entries with bad data were neglected in the evaluation. Wind speed multipliers for the MSSV release were selected based upon the stability class for each hour of data. The wind speed multiplier and the 10 m wind speed were multiplied together to obtain the wind speed at the height of the release for each hour of data.

The wind speed multiplier is selected based upon the stability class, and is taken from ARCON96 case runs. The results of these case runs present the wind speed correction factors for the MSSV release location.

The valid hourly release height wind speeds were then utilized to determine the 95th percentile value. The 95th percentile wind speed for the Unit 4 MSSV release height is 24.6 miles per hour. That is, 95% of all of the hourly wind speeds at the MSSV release height are less than 24.6 miles per hour.

The minimum exit velocity for the MSSVs and ADVs at the opening setpoint are greater than 5 times the above listed 95th percentile wind speed for the duration of the steam release through the valves. The MSSV exit velocity is greater than 125.7 ft/s and the nominal ADV exit velocity is 194.0 ft/s. For these conditions, the plume rise credit discussed in Section 6 of Reg. Guide 1.194 may be applied. This factor of 5 reduction to the χ/Q values for the MSSV and ADV releases was only credited prior to the beginning of RCS cooldown. This conservatively neglects any plume rise credit which may be applicable during the early portion of the cooldown.

1.9 Sump pH Determination

Standard Review Plan Section 6.5.2 requires that the pH of the solution collected in the containment sump after completion of injection of containment spray, ECCS water, and all additives be maintained at a level sufficiently high to provide assurance that significant long-term iodine re-evolution does not occur. The SRP identifies that long-term iodine retention may be assumed only when the equilibrium sump solution pH, after mixing and dilution with the primary coolant and ECCS injection is above 7; and should be achieved by the onset of the spray recirculation mode.

A manual calculation was performed to determine the amount of sodium tetraborate decahydrate that must be located in the containment sump to ensure a pH of at least 7.0 by the time of the onset of containment spray. This calculation used conservative assumptions for sump fill rate, RWST drain rate, RCS volume, RWST transfer volume, sump liquid temperature, and fluid hold-up in containment. Sodium tetraborate decahydrate dissolution rates were based upon information contained in WCAP-16596-NP (Reference 5.33). This analysis also considered the formation of nitric acid due to the irradiation of water in the sump and the generation of hydrochloric acid due to the radiolysis of electrical cable insulation per the guidelines of NUREG/CR-5950 (Reference 5.30). The NaTB will be placed into two large and eight small baskets. The large baskets will have a length and width of 4.5

feet and a height of 2.77 feet. Small baskets will be 3 feet square at the base and are 2.5 feet high. A parametric study was performed to determine the minimum and maximum NaTB loading of the baskets. Lower NaTB levels reduce the total amount of buffering agent available for dissolution. Higher NaTB levels require higher sump liquid levels before the top surface area of the NaTB in the baskets become exposed to water. The amount of buffering agent required and the resulting pH values are presented in Table 1.9-1.

2.0 Radiological Consequences – Event Analyses

2.1 Loss of Coolant Accident (LOCA)

2.1.1 Background

This event is assumed to be caused by an abrupt failure of the main reactor coolant pipe and the ECCS fails to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity is released from the containment and from there, released to the environment by means of containment leakage and leakage from the ECCS. This event is described in the Section 14.3.5 of the UFSAR.

2.1.2 Compliance with Reg. Guide 1.183 Regulatory Positions

The LOCA dose consequence analysis is consistent with the guidance provided in Reg. Guide 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on Reg. Guide 1.183, Regulatory Position 3.1. A conservative power level is used which exceeds 102% of the rated core thermal power. The resulting core source term is provided in Table 1.7.4-1. The core inventory release fractions for the gap release and early in-vessel damage phases of the LOCA are consistent with Regulatory Position 3.2 and Table 2 of Reg. Guide 1.183.

2. Regulatory Position 2 - The sump pH is controlled at a value greater than 7.0 based on the addition of sodium tetraborate decahydrate baskets. Therefore, the chemical form of the radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form.

3. Regulatory Position 3.1 - The activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase.

4. Regulatory Position 3.2 - Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 5.58 hr^{-1}. This removal is credited in both the sprayed and unsprayed regions of containment.

   A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region of containment as well as in the sprayed region when sprays are not operating. Industry Degraded

Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," December 1983, documents results from Containment Systems Experiment testing. These tests show that settling of aerosols due to gravity is the dominant natural mechanism for fission product retention. This report finds that significant removal by sedimentation would be expected even at very low particulate concentrations. Figure 4-2 of IDCOR Program Technical Report 11.3 shows a ten-fold reduction in the airborne cesium concentration over a 7-hour period at relatively low concentrations. This represents an aerosol removal rate of 0.33 hr^{-1}. A more conservative value of 0.1 hr^{-1} is used in the analyses based upon NRC approval of this value in the safety evaluations for the St. Lucie Unit 2 License Amendment No. 152 in September 2008 (ADAMS Accession No. ML082060400) and the Palisades Nuclear Plant License Amendment No. 226 in September 2007 (ADAMS Accession No. ML72470667).

No removal of organic iodine by natural deposition is assumed.

5. Regulatory Position 3.3 – A single train of containment spray provides coverage to 34.5% of the containment volume. Therefore, the Turkey Point containment building atmosphere is not considered to be a single, well-mixed volume. The containment is divided into three regions: sprayed, an unsprayed region above the operating deck and an unsprayed region below the operating deck. The mixing rates for the containment sprayed and unsprayed regions are based on a GOTHIC analysis which produced results consistent with NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings". Precedent for similar containment mixing results is established in the Fort Calhoun Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 198 to DPR-40 issued April 4, 2001.

   The GOTHIC analysis utilized for Turkey Point to demonstrate the level of spray induced mixing in containment included both subdivided and lumped parameter models. The detailed subdivided models were used to calculate flow patterns produced by the containment sprays and the emergency containment coolers. Gas concentrations from the subdivided models were compared with concentrations in the lumped parameter model and used to determine equivalent mixing flow rates for the lumped model.

   Based on the results of this analysis, the AST dose calculations were conducted using a three volume model similar to the lumped parameter GOTHIC model. The AST model includes separate volumes representing the unsprayed lower, unsprayed upper and sprayed upper regions of containment. Mixing flow rates up to 375,000 cfm between lower and upper unsprayed regions and 990,000 cfm between upper sprayed and unsprayed regions conservatively cover the possible combinations of sprays and emergency fans that may be available during an accident scenario.

   The method used in the Turkey Point AST LOCA analysis for determining the time period required to reach an elemental iodine DF of 200 was based on a containment atmosphere peak iodine concentration equal to 40 percent of the core iodine inventory per Table 2 of Reg. Guide 1.183.

   The SRP requires that the elemental iodine spray removal coefficient should be set to zero when a decontamination factor (DF) of 200 is reached for elemental iodine. In addition, the particulate spray removal coefficient should be reduced by a factor of 10 when a DF of 50 is reached for the aerosols.

   As discussed in the SRP, the iodine decontamination factor (DF) is a function of the effective iodine partition coefficient between the sump and containment atmosphere. Thus, the loss of iodine due to other mechanisms (containment leakage, surface deposition, etc.), would not be included in the determination of the time required to reach a DF of 200. In addition, since the iodine in the containment atmosphere and sump are decaying at the same rate, decay should not be included in determining the time to reach a DF of 200. Additional RADTRAD-NAI cases were performed for determining the time to reach a decontamination factor of 200.

   The first RADTRAD-NAI case was used to determine the peak containment atmosphere elemental iodine concentration and amount of aerosol in the containment atmosphere. This case included:

- No containment spray
- No elemental iodine surface deposition
- No aerosol surface deposition
- No decay
- No containment leakage

The second RADTRAD-NAI case determined the time required to reach a DF of 200 based on the peak elemental iodine concentration from the first RADTRAD-NAI case. The second RADTRAD-NAI case included:

- Containment sprays actuated at 0.018 hours
- No surface deposition
- No decay
- No containment leakage

Due to the high mixing rate between the containment regions, the activity in all three containment regions was considered. The second RADTRAD-NAI case showed that a DF of 200 for elemental iodine was reached at a time greater than 2.305 hours.

A separate RADTRAD-NAI case was then used to determine the time required to reach a DF of 50 for aerosols based on the peak aerosol mass from the first RADTRAD-NAI case. This RADTRAD-NAI case included:

- Containment sprays actuated at 0.018 hours
- Aerosol surface deposition credited
- No decay
- No containment leakage

Due to the high mixing rate between the containment regions, the activity in all three containment regions is considered. The third RADTRAD-NAI case showed that a DF of 50 was reached at a time greater than 3.06 hours.

Containment spray flow is assumed to be stopped for a period of five minutes to allow for manual re-alignment of the pump suction from the RWST to the recirculation sump. Termination of spray flow is considered in the determination of the iodine decontamination factors and is reflected in the mixing rates between the containment regions

6. Regulatory Position 3.4 - Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems is not credited in this analysis.

7. Regulatory Position 3.5 - This position relates to suppression pool scrubbing in BWRs, which is not applicable to Turkey Point.

8. Regulatory Position 3.6 - This position relates to activity retention in ice condensers, which is not applicable to Turkey Point.

9. Regulatory Position 3.7 - A containment leak rate of 0.20% per day of the containment air is assumed for the first 24 hours based on proposed Technical Specification 6.8.4h. After 24 hours, the containment leak rate is reduced to 0.10% per day of the containment air. The containment leakage was applied to all three containment regions.

10. Regulatory Position 3.8 - Routine containment purge is considered in this analysis. The purge release evaluation assumes that 100% of the radionuclide inventory of the RCS is released instantaneously at the beginning of the event. The containment purge flow is 7000 cfm and is isolated after 8 seconds, which is before the onset of the gap release phase. No filters are credited.

11. Regulatory Position 4.1 through 4.6 provide guidance for the evaluation of the transport, reduction, and release of radioactive material through dual containment structures. These positions are not applicable to Turkey Point.

12. Regulatory Position 5.1 - Engineered Safety Feature (ESF) systems that recirculate water outside the primary containment are assumed to leak during their intended operation. With the exception of noble gases, all fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core.

13. Regulatory Position 5.2 - Leakage from the ESF system is taken as two times the value from UFSAR Table 6.2-12. ECCS leakage is assumed to start at the earliest time the recirculation flow occurs in these systems and continue for the 30-day duration. Backleakage to the Refueling Water Storage Tank is also considered separately as 0.1 gph, which exceeds two times the expected leakage through the two sets of isolation valves between the RWST and recirculation flow. Backleakage to the RWST is assumed to begin at the start of recirculation and continue for the remainder of the 30-day duration.

14. Regulatory Position 5.3 - With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

15. Regulatory Position 5.4 – A flashing fraction of 9.2% was calculated based on a conservative maximum sump liquid temperature and containment design pressure. However, consistent with Regulatory Position 5.5, the flashing fraction for ECCS leakage is assumed to be 10%. For ECCS leakage back to the RWST, the analysis demonstrates that the temperature of the leaked fluid will cool below 212°F prior to release to the RWST tank.

16. Regulatory Position 5.5 - The amount of iodine that becomes airborne is conservatively assumed to be 10% of the total iodine activity in the leaked fluid for the ECCS leakage entering the Reactor Auxiliary Building. For the ECCS leakage back to the RWST, the sump and pH history and temperature are used to evaluate the amount of iodine that enters the RWST air space.

17. Regulatory Position 5.6 – For ECCS leakage into the auxiliary building, the form of the released iodine is 97% elemental and 3% organic. No credit for ESF filtration of the ECCS leakage nor holdup or dilution in the auxiliary building is taken. For releases from the RWST, the temperature and pH history of the sump and RWST are considered in determining the radioiodine available for release and the chemical form. Credit is taken for dilution of activity in the RWST.

18. Regulatory Position 6 - This position relates to MSSV leakage in BWRs, which is not applicable to Turkey Point.

19. Regulatory Position 7 - Containment purge is not considered as a means of combustible gas or pressure control in this analysis; however, the effect of routine containment purge before isolation is considered.

2.1.3 Methodology

For this event, the Control Room ventilation system cycles through both modes of operation (the operational modes are summarized in Table 1.6.3-1). Inputs and assumptions fall into three main categories: Radionuclide Release Inputs, Radionuclide Transport Inputs, and Radionuclide Removal Inputs.

For the purposes of the LOCA analyses, a major LOCA is defined as a rupture of the RCS piping, including the double-ended rupture of the largest piping in the RCS, or of any line connected to that system up to the first closed valve. Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer low-pressure trip setpoint is reached. A safety injection system signal is actuated when the appropriate setpoint (high containment pressure) is reached. The following measures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat, and

2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

Release Inputs

The core inventory of the radionuclide groups utilized for this event is based on Reg. Guide 1.183, Regulatory Position 3.1, at 2652 MW_{th} and is provided as Table 1.7.4-1. The source term represents end of cycle conditions assuming enveloping initial fuel enrichment and an average core burnup of 45,000 MWD/MTU.

Per TS 3.6.1.2 and proposed TS 6.8.4h, the leakage rate acceptance criteria for the containment is 0.20% of the containment air weight per day. Therefore, for the first 24 hours, the containment is assumed to leak at a rate of 0.20% of the containment air per day. Per Reg. Guide 1.183, Regulatory Position 3.7, the primary containment leakage rate is reduced by 50% at 24 hours into the LOCA to 0.10% /day based on the post-LOCA primary containment pressure history.

A containment purge is also assumed coincident with the beginning of the LOCA. Since the purge is isolated prior to the initial release of fission products from the core at 30 seconds, only the initial RCS activity is available for release via this pathway. The release of 7000 cfm is modeled for 8 seconds until isolation occurs.

The ECCS leakage to the auxiliary building is 4,650 cc/hr based upon two times the current licensing basis value of 2,325 cc/hr. The leakage is assumed to start at 15 minutes into the event and continue throughout the 30-day period. This portion of the analysis assumes that 10% of the total iodine is released from the leaked liquid. The form of the released iodine is 97% elemental and 3% organic.

The ECCS backleakage to the RWST is assumed to be 0.1 gph based upon doubling of the expected total seat leakage through both sets of motor operated valves which isolate the recirculation flow from the RWST. The leakage is assumed to start at 15 minutes into the event when recirculation begins and continue throughout the 30-day period. Note that based on the leakage rate and the size of the piping, the leakage would not reach the RWST for an extended period of time after recirculation begins. This time period is conservatively not credited for determining when the leakage reaches the RWST (i.e., the leakage is assumed to reach the RWST instantaneously allowing no time for radioactive decay); however, this time period is credited for determining the temperature of the leakage reaching the RWST. It should also be noted that based on the small leak rate and pipe length between the ECCS isolation valves and the RWST, the leaked ECCS fluid would probably not reach the RWST during the first 30 days of the event.
Based on sump pH history, the iodine in the sump solution is assumed to all be in nonvolatile form. However, when introduced into the acidic solution of the RWST inventory, there is a potential for the particulate iodine to convert into the elemental form. The fraction of the total iodine in the RWST which becomes elemental is both a function of the RWST pH and the total iodine concentration. The amount of elemental iodine in the RWST fluid which then enters the RWST air space is a function of the temperature-dependent iodine partition coefficient.

The time-dependent concentration of the total iodine in the RWST, including stable iodine, was determined from the tank liquid volume and leak rate. This iodine concentration ranged from a minimum value of 0.0 at the beginning of the event to a maximum value of 8.03E-08 gm-atom/liter at 30 days (see Table 2.1-3). Based on these results, a constant value of 1.0E-07 gm-atom/liter is applied in the analysis. Due to the small backleakage rate, the RWST pH remains at the conservative initial value of 3.0 for the duration of the event. Using this pH and the time-dependent total iodine concentration in the RWST liquid space, the amount of iodine which is converted to the elemental form was determined using guidance provided in NUREG/CR-5950. This RWST elemental iodine fraction ranged from 0.0 at the beginning of the event to a maximum of 0.0882 (see Table 2.1-4). Conservative application of the constant total iodine concentration of 1.0E-07 gm-atom/liter resulted in an elemental iodine fraction of 0.1058, which was then used to calculate the iodine release rate from the RWST.

The elemental iodine in the liquid leaked into the RWST is assumed to become volatile and partition between the liquid and vapor space in the RWST based upon the temperature dependent partition coefficient for elemental iodine as presented in NUREG-5950. The RWST is a vented tank; therefore, there will be no pressure transient in the air region that would affect the partition coefficient. Since no boiling occurs in the RWST, the release of the activity from the vapor space within the RWST is calculated based upon the displacement of air by the incoming leakage and the expansion due to the daily heating and cooling cycle of the contents of the RWST. The average daily temperature swing of 10.1 °F is applied for every 24-hour period for 30 days and no credit is taken for cooling of the tank contents via conduction. The iodine release is implemented via an adjustment to the vapor flow rate from the RWST. This adjustment accounts for the time-dependent relationship between the elemental iodine concentration in the RWST vapor space with respect to the sump iodine concentration. The average adjusted RWST vapor release rate is then applied to the entire iodine inventory in the containment sump.

This same approach is used with the organic iodine. An organic iodine fraction of 0.0015 is assumed in combination with a partition coefficient of 1.0. The particulate portion of the leakage is assumed to be retained in the liquid phase of the RWST. Therefore, the total iodine flow from the RWST represents the sum of the elemental and organic concentrations in the RWST vapor space. The average adjusted RWST release rate is presented in Table 2.1-5.

The release points for each of the above sources is provided in Table 1.8.1-3.

Transport Inputs

During the LOCA event, the activity collected in containment is assumed to be released to the environment via a ground level release from the containment building. The containment purge activity is modeled as a ground level release via the plant stack with no filtration. The activity from ECCS components and from RWST leakage are modeled as an unfiltered ground level releases from the location of the RWST. For the ECCS leakage, the X/Qs from the RWST to the emergency intakes are more limiting than from any of the Auxiliary Building penetrations.

The Control Room atmospheric dispersion factors (*X/Qs*) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intakes. Onsite *X/Q* values for the various combinations of release points and receptor locations are presented in Table 1.8.1-2. Table 1.8.1-3 presents the Release-Receptor pairs applicable to the control room dose from the LOCA release points for the different modes of control room operation during the event.

For the EAB dose analysis, the *X/Q* factor corresponding to the 0-2 hour time period was used for the entire duration of the event. The LPZ dose is determined using the *X/Q* factors for the appropriate time intervals also. These *X/Q* factors are provided in Table 1.8.2-1.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air intake to the Control Room during this mode is 1000 cfm of unfiltered fresh air.

- Control Room isolation will occur from an SI actuation or high radiation in containment or at the normal intake. A 30-second delay is conservatively applied to account for the time to reach the signal, the diesel generator start time, load sequencing and damper actuation and positioning time. After isolation of the Control Room normal air intake, the air flow distribution consists of 525 cfm of filtered makeup flow through the worst of the two emergency intakes, 115 cfm of unfiltered inleakage and 375 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 97.5% for particulates, elemental, and organic iodine.

<u>Removal Inputs</u>

Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 5.58 hr^{-1}. This removal is credited in the sprayed and unsprayed regions. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed regions. No natural deposition removal of aerosols is credited in the sprayed region. No removal of organic iodine by natural deposition is assumed.

Containment spray provides coverage to 534,442 ft^3 of the total 1.55E6 ft^3 containment volume. Therefore, the Turkey Point containment building atmosphere is not considered to be a single, well-mixed volume. The containment is divided into three regions: sprayed and unsprayed regions above the operating deck and an unsprayed region below the operating deck. The mixing rates between the regions are based on a separate sensitivity study evaluating various combinations of containment fans and sprays to produce the most conservative mixing rates. The final conservative mixing rates are 990,000 cfm between the upper sprayed and upper unsprayed containment regions above the operating deck and 375,000 cfm between the lower unsprayed region below the operating deck and the upper unsprayed region above the operating deck.

According to SRP 6.5.2, the effectiveness of elemental iodine removal by the containment sprays is presumed to end when the decontamination factor (DF) reaches a maximum value of 200. The maximum initial airborne elemental iodine concentration is based on the release of 40 percent of the core iodine inventory. With the elemental iodine spray removal rate set to the SRP limit of 20 hr^{-1}, the decontamination factor for elemental iodine reaches 200 at just over 2.305 hours.

The spray aerosol removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated aerosol iodine removal rate of 6.44 hr^{-1}, the time for containment spray to produce an aerosol decontamination factor of 50 is calculated to be greater than 3.06 hours.

Filter removal in the Control Room Emergency Mode is simulated using conservative assumptions based on plant design data as listed in Table 1.6.3-1.

2.1.4 Radiological Consequences

The radiological consequences of the design basis LOCA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. In addition, the MicroShield code, is used to develop direct shine doses to the Control Room. The post accident doses are the result of four distinct activity releases:

1. Containment leakage.

2. ESF system leakage into the Auxiliary Building.

3. ESF system leakage into the RWST.

4. Containment Purge at event initiation.

The dose to the Control Room occupants includes terms for:

1. Contamination of the Control Room atmosphere by intake and infiltration of radioactive material from the containment and from ECCS leakage.

2. External radioactive plume shine contribution from the containment and ECCS leakage releases. This term takes credit for Control Room structural shielding.

3. A direct shine dose contribution from the Containment's contained accident activity. This term takes credit for both Containment and Control Room structural shielding.

4. A direct shine dose contribution from the activity collected on the Control Room ventilation filters.

As shown in Table 2.1-6, the sum of the results of all dose contributions for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.2 Fuel Handling Accident (FHA)

2.2.1 Background

This event consists of the drop of a single fuel assembly either in the Fuel Handling Building (FHB) or inside of Containment. The FHA is described in Section 14.2.1.2 of the UFSAR. The UFSAR description of the FHA specifies a case that assumes all of the fuel rods in a single fuel assembly are damaged.

This analysis considers both a dropped fuel assembly inside the containment with the equipment hatch open, and an assembly dropped inside the FHB without credit for filtration of the Fuel Handling Building exhaust. The source term released from the overlying water pool is the same for both the FHB and the containment cases. Reg. Guide 1.183 imposes the same 2-hour criteria for the direct unfiltered release of the activity to the environment for either location.

A minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and FHB release locations. This water level ensures an elemental iodine decontamination factor of 285 per the guidance provided in NRC Regulatory Issue Summary 2006-04.

2.2.2 Compliance with Reg. Guide 1.183 Regulatory Positions

The FHA dose consequence analysis is consistent with the guidance provided in Reg. Guide 1.183 Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," as discussed below:

1. Regulatory Position 1.1 - The amount of fuel damage is assumed to be all of the fuel rods in a single fuel assembly per UFSAR Section 14.2.1.2.

2. Regulatory Position 1.2 - The fission product release from the breached fuel is based on Regulatory Positions 3.1 and 3.2 of Reg. Guide 1.183. Section 1.7.5 provides a discussion of how the FHA source term is developed. A listing of the FHA source term is provided in Table 1.7.5-1. The gap activity available for release is modified from that specified by Table 3 of Reg. Guide 1.183 to account for high burnup fuel as described in Section 1.7.6. This activity is assumed to be released from the fuel assembly instantaneously.

3. Regulatory Position 1.3 - The chemical form of radioiodine released from the damaged fuel into the spent fuel pool is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The cesium iodide is assumed to completely dissociate in the spent fuel pool resulting in a final iodine distribution of 99.85% elemental iodine and 0.15% organic iodine.

4. Regulatory Position 2 - A minimum water depth of 23 feet is maintained above the damaged fuel assembly. Therefore, a decontamination factor of 285 is applied to the elemental iodine and a decontamination factor of 1 is applied to the organic iodine. As a result, the breakdown of the iodine species above the surface of the water is 57% elemental and 43% organic. Guidance for the use of 285 for the elemental iodine decontamination factor is provided in NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms."

5. Regulatory Position 3 - All of the noble gas released is assumed to exit the pool without mitigation. All of the non-iodine particulate nuclides are assumed to be retained by the pool water.

6. Regulatory Position 4.1 - The analysis models the release to the environment over a 2-hour period.

7. Regulatory Position 4.2 - No credit is taken for filtration of the release.

8. Regulatory Position 4.3 - No credit is taken for dilution of the release.

9. Regulatory Position 5.1 - The containment equipment hatch is assumed to be open at the time of the fuel handling accident.

10. Regulatory Position 5.2 - No automatic isolation of the containment is assumed for the FHA.

11. Regulatory Position 5.3 - The release from the fuel pool is assumed to leak to the environment over a two-hour period.

12. Regulatory Position 5.4 - No ESF filtration of the containment release is credited.

13. Regulatory Position 5.5 - No credit is taken for dilution or mixing in the containment atmosphere.

2.2.3 Methodology

The input assumptions used in the analysis of the FHA dose consequences are provided in Table 2.2-1. It is assumed that the fuel handling accident occurs at 72 hours after shutdown of the reactor per TS LCO 3.9.3. 100% of the gap activity is assumed to be instantaneously released from a single fuel assembly into the fuel pool. A minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event. All of the noble gas released from the damaged fuel assembly is assumed to escape from the pool. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the pool liquid. The iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. The activity released from the pool is then assumed to leak to the environment over a two-hour period. No credit for dilution in the containment or FHB is taken.

The FHA source term meets the requirements of Regulatory Position 1 of Appendix B to Reg. Guide 1.183. Gap release fractions have been increased to address fuel with rod average burnups greater than 54,000 MWD/MTU. The FHA source term is listed in Table 1.7.5-1 and the applicable gap release fractions are presented in Section 1.7.6.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air.

- For the FHA in containment, Control Room isolation occurs on high radiation on the containment radiation monitors. A 30 second delay time is assumed for signal processing and damper closure. For the release from the Fuel Handling Building, the Control Room is assumed to be manually isolated by the operators 30 minutes after the beginning of the event. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 115 cfm of unfiltered inleakage, and 375 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 97.5% for particulate, elemental iodine, and organic iodine.

The atmospheric dispersion factors (*X/Qs*) used for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3. The Control Room atmospheric dispersion factors applied to the FHA in containment are based upon a release from the most limiting containment equipment/personnel hatch. For the Fuel Handling Building release, the most limiting X/Qs correspond to a release from the Unit 4 spent fuel pool.

The EAB and LPZ doses are determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are provided in Table 1.8.2-1.

2.2.4 Radiological Consequences

The radiological consequences of the FHA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.2-2 the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.3 Main Steamline Break (MSLB)

2.3.1 Background

This event consists of a double-ended break of one main steam line outside of containment. The radiological consequences of such an accident bound those of a MSLB inside containment. The faulted steam generator rapidly depressurizes and releases the initial contents of the steam generator secondary to the environment. Plant cool down is achieved via the remaining unaffected steam generators. This event is described in UFSAR Section 14.2.5.

2.3.2 Compliance with Reg. Guide 1.183 Regulatory Positions

The analysis of the MSLB dose consequences is consistent with the guidance provided in Reg. Guide 1.183, Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," as discussed below:

1. Regulatory Position 1 – No fuel damage is postulated to occur for the Turkey Point MSLB event.

2. Regulatory Position 2 – No fuel damage is postulated to occur for the Turkey Point MSLB event. Therefore, two cases of iodine spiking are evaluated.

3. Regulatory Position 2.1 - One iodine spiking case assumes a reactor transient prior to the postulated MSLB that raises the primary coolant iodine concentration to the maximum allowed by TS 3.4.8 Figure 3.4-1, which is a value of 60.0 μCi/gm DE I-131. This is the pre-accident spike case.

4. Regulatory Position 2.2 - One case assumes the transient associated with the MSLB causes an iodine

spike. The spiking model assumes the primary coolant activity is initially at the proposed TS 3.4.8 value of 0.25 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.

5. Regulatory Position 3 - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

6. Regulatory Position 4 - Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

7. Regulatory Position 5.1 - The primary-to-secondary leak rate is equal to the value specified by proposed TS 6.8.4.j.b.2, which is 0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions.

8. Regulatory Position 5.2 - The density used in converting primary-to-secondary volumetric leak rates to mass leak rates is 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

9. Regulatory Position 5.3 – The primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F. This is conservatively calculated to occur at 125.4 hours. The release of radioactivity from the unaffected steam generators is conservatively assumed to continue until RHR is capable of removing decay heat and for providing for any further cooldown, which occurs at 63 hours.

10. Regulatory Position 5.4 - All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

11. Regulatory Position 5.5.1 - In the faulted steam generator, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. For the unaffected steam generators used for plant cooldown, a portion of the leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

12. Regulatory Position 5.5.2 - Any postulated leakage that immediately flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the bulk water is credited.

13. Regulatory Position 5.5.3 - All leakage that does not immediately flash is assumed to mix with the bulk water.

14. Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected steam generators is limited by the moisture carryover. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the steam generator carryover rate of less than 1%. No reduction in the release is assumed from the faulted steam generator.

15. Regulatory Position 5.6 - Steam generator tube bundle uncovery in the intact steam generators is postulated for up to 30 minutes following a reactor trip. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing

fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.3.3 Other Assumptions

1. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

2. The steam mass release rates for the intact steam generators are provided in Table 2.3-2.

3. Data used to calculate the iodine equilibrium appearance rate are provided in Table 2.3-4, "Iodine Equilibrium Appearance Assumptions."

4. This evaluation assumes that the RCS mass remains constant throughout the MSLB event. No change in the RCS mass is assumed as a result of the primary-to-secondary leakage or from the safety injection system.

5. Radionuclide concentrations in the secondary side fluid of the steam generators assume that Auxiliary Feedwater is provided to maintain a constant secondary mass during periods of steam release.

6. Releases from the faulted main steam line are postulated to occur from the main steam line associated with the most limiting atmospheric dispersion factors. Releases from the unaffected steam generators are postulated to occur from the MSSV or ADV with the most limiting atmospheric dispersion factors.

7. The steam generator partition factor for iodine and particulates is applied in the analysis by reducing the steam release rate from the steam generator compartment. This methodology conservatively allows the activity which is not released to remain in the steam generator compartment and contribute to the radionuclide concentration.

2.3.4 Methodology

Input assumptions used in the dose consequence analysis of the MSLB are provided in Table 2.3-1. The postulated accident assumes a double-ended break of one main steam line outside containment. The radiological consequences of such an accident bound those of a MSLB inside of containment. Upon a MSLB, the faulted steam generator rapidly depressurizes and releases the initial contents to the environment. Plant cooldown is achieved via the remaining unaffected steam generators.

The analysis assumes that activity is released as reactor coolant enters the steam generators due to primary-to-secondary leakage. The source term for this activity is presented in Table 1.7.2-1. All noble gases associated with this leakage are assumed to be released directly to the environment. Primary-to-secondary leakage into the faulted steam generator is also assumed to directly enter the atmosphere. Leakage into the unaffected steam generators is partitioned by the secondary fluid and released via the MSSVs and ADVs. All primary-to-secondary leakage is assumed to continue until the primary system is cooled to 212 °F at 125.4 hours. The release of the initial iodine content of the steam generator secondary is also considered. The source term for this activity is presented in Table 1.7.3-1.

Fuel damage is not postulated for the MSLB event. Consistent with Regulatory Guide 1.183, Appendix E, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity released is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike; and (2) maximum accident-induced or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated MSLB event. The primary coolant iodine concentration is increased to the maximum value of

60 μCi/gm DE I-131 permitted by TS 3.4.8. The iodine activities for the pre-accident spike case are presented in Table 2.3-3.

For the case of the accident-induced spike, the postulated MSLB event induces an iodine spike. The RCS activity is initially assumed to be 0.25 μCi/gm DE I-131 as allowed by proposed TS 3.4.8. Iodine is released from the fuel into the RCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours. With iodine activity at equilibrium, the iodine release rate is equal to the rate at which iodine is lost due to decay, purification, and primary system leakage. Parameters used in the determination of the iodine equilibrium release rate are provided in Table 2.3-4. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 2.3-5.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air.

- Control Room is isolated following receipt of a safety injection signal. A 41.5-second delay is applied to account for the signal processing and damper closure time. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 115 cfm of unfiltered inleakage, and 375 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 97.5% for particulate, elemental iodine, and organic iodine.

The atmospheric dispersion factors (*X/Qs*) used for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3. Releases from the intact steam generators are assumed to occur from the MSSV/ADV which produces the most limiting *X/Q*. Releases from the faulted steam generator are assumed to occur from the location on a steam line closest the in-service intake.

The EAB and LPZ doses are determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are provided in Table 1.8.2-1.

2.3.5 Radiological Consequences

The radiological consequences of the MSLB Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Cases for MSLB pre-accident and concurrent iodine spikes are analyzed. As shown in Table 2.3-6, the results of both cases for EAB dose, LPZ dose, and Control Room dose are within the appropriate regulatory acceptance criteria.

2.4 Steam Generator Tube Rupture (SGTR)

2.4.1 Background

This event is assumed to be caused by the instantaneous rupture of a Steam Generator tube that relieves to the lower pressure secondary system. No melt or clad breach is postulated for the Turkey Point SGTR event. This event is described in UFSAR Section 14.2.4.

2.4.2 Compliance with Reg. Guide 1.183 Regulatory Positions

The SGTR dose consequence analysis is consistent with the guidance provided in Reg. Guide 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," as discussed below:

1. Regulatory Position 1 - No fuel damage is postulated to occur for the Turkey Point SGTR event.

2. Regulatory Position 2 - No fuel damage is postulated to occur for the Turkey Point SGTR event. Two cases of iodine spiking are assumed.

3. Regulatory Position 2.1 - One case assumes a reactor transient prior to the postulated SGTR that raises the primary coolant iodine concentration to the maximum allowed by TS 3.4.8 Figure 3.4-1, which is a value of 60.0 μCi/gm DE I-131 for the analyzed conditions. This is the pre-accident spike case.

4. Regulatory Position 2.2 - One case assumes the transient associated with the SGTR causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the proposed TS 3.4.8 value of 0.25 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case.

5. Regulatory Position 3 - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

6. Regulatory Position 4 - Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

7. Regulatory Position 5.1 - The primary-to-secondary leak rate is equal to the value specified by proposed TS 6.8.4.j.b.2, which is 0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions.

8. Regulatory Position 5.2 - The density used in converting primary-to-secondary volumetric leak rates to mass leak rates is 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

9. Regulatory Position 5.3 - The primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F. This is conservatively calculated to occur at 125.4 hours. The release of radioactivity from the unaffected steam generators is assumed to continue until RHR is capable of removing decay heat and for providing for any further cooldown, which occurs at 63 hours. Termination of the ruptured steam generator activity release is occurs when the ruptured steam generator is isolated at 30 minutes by operator action. While this isolation terminates releases from the ruptured steam generator, primary-to-secondary leakage continues to provide activity for release from the unaffected steam generators.

10. Regulatory Position 5.4 - The release of fission products from the secondary system is evaluated with the assumption of a loss of offsite power coincident with reactor trip.

11. Regulatory Position 5.5 - All noble gases released from the primary system are assumed to be released to the environment without reduction or mitigation.

12. Regulatory Position 5.6 - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:

    - Appendix E, Regulatory Position 5.5.1 - A portion of the primary-to-secondary ruptured tube flow following the SGTR is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary. The flashed flow is released to the environment with no mitigation. For the unaffected steam generators, flashing is considered immediately following plant trip when tube uncovery is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

    - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the bulk water is credited.

    - Appendix E, Regulatory Position 5.5.3 - All of the steam generator leakage and ruptured tube flow that does not immediately flash is assumed to mix with the bulk water.

    - Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the steam generator carryover rate of less than 1%.

    - Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery in the unaffected steam generators is postulated for up to 30 minutes following a reactor trip. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.4.3 Other Assumptions

1. RCS and steam generator volumes are assumed to remain constant throughout both the pre-accident and the accident-induced iodine spike SGTR events.

2. During a SGTR event, from the onset of the tube rupture until the time of reactor trip, there will be no actual steam releases from through ADVs or MSSVs. Radionuclides will most likely enter the atmosphere through the condenser Steam Jet Air Ejector (SJAE). Due to relative proximity of the SJAE to the normal Control Room intake, this release-receptor pair produces a more limiting atmospheric dispersion factor than releases from the ADVs/MSSVs to the normal intake. In addition, the pre-trip flashing fraction of the primary-to-secondary leakage is substantially higher than the pos-trip flashing fraction. For these reasons, the analysis assumes full rated steam flow from the steam generators prior to reactor trip. In addition to the partition factor of 100 in the steam generator, a partition factor of 100 is also applied to iodines and particulates released from the SJAE prior to the reactor trip. This condenser partition factor is no longer used when the steam release from the ADVs/MSSVs begins. The value of 100 for the condenser partition factor is consistent with that approved for use by Kewaunee and South Texas.

3. The steam generator and condenser partition factors for iodine and particulates are applied in the analysis by reducing the steam release rate from the steam generator compartment. This methodology conservatively allows the activity which is not released to remain in the steam generator compartment and contribute to the radionuclide concentration.

4. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

5. Radionuclide concentrations in the secondary side fluid of the steam generators assume that Auxiliary Feedwater is provided to maintain a constant secondary mass during periods of steam release.

6. The steam release rates and ruptured tube flow rates are provided in Table 2.4-2.

7. Data used to calculate the iodine equilibrium appearance rate are provided in Table 2.4-4, "Iodine Equilibrium Appearance Assumptions."

2.4.4 Methodology

Input assumptions used in the dose consequence analysis of the SGTR event are provided in Table 2.4-1. This event is assumed to be caused by the instantaneous rupture of a steam generator tube releasing primary coolant to the lower pressure secondary system. Initial radionuclide releases occur through the condenser SJAE until the time of reactor trip, thereby causing steam relief directly to the atmosphere from the ADVs or MSSVs. This direct steam relief continues until the ruptured steam generator is isolated at 30 minutes.

A thermal-hydraulic analysis is performed to determine a conservative maximum break flow, break flashing flow, and steam release inventory through the ruptured steam generator relief valves. The analysis assumes that activity is released as reactor coolant enters the steam generators due to primary-to-secondary leakage. The source term for this activity is presented in Table 1.7.2-1. All noble gases associated with this leakage are assumed to be released directly to the environment. Primary coolant is released into the ruptured steam generator through the ruptured tube and from a fraction of the total proposed allowable primary-to-secondary leakage until the ruptured steam generator is isolated at 30 minutes. Additional activity, based on the proposed primary-to-secondary leakage limits, is released via the unaffected steam generators. All primary-to-secondary leakage is assumed to continue until the temperature of the leakage is less than 212°F, which is conservatively calculated to occur at 125.4 hours. Steam release from the unaffected steam generators is assumed to continue until RHR is capable of removing decay heat and for providing for any further cooldown, which occurs at 63 hours. The release of the initial iodine content of the steam generator secondary is also considered. The source term for this activity is presented in Table 1.7.3-1.

Per the Turkey Point UFSAR, Section 14.2.4, no fuel melt or clad breach is postulated for the SGTR event. Consistent with Reg. Guide 1.183 Appendix F, Regulatory Position 2, if no or minimal fuel damage is postulated for the limiting event, the activity release is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike, and (2) maximum accident-induced, or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated SGTR event. The primary coolant iodine concentration is increased to the maximum value of 60 μCi/gm DE I-131 permitted by TS 3.4.8. The iodine activities for the pre-accident spike case are presented in Table 2.4-3.

For the case of the accident-induced spike, the postulated STGR event induces an iodine spike. The RCS activity is initially assumed to be 0.25 μCi/gm DE I-131 as allowed by proposed TS 3.4.8. Iodine is released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. With iodine activity at equilibrium, the iodine release rate is equal to the rate at which iodine is lost due to decay, purification, and primary system leakage. Parameters used in the determination of the iodine equilibrium release rate are provided in Table 2.4-4. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 2.4-5. All other release assumptions for this case are identical to those for the pre-accident spike case.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air.
- Control Room is isolated due to a safety injection, which occurs at 291 seconds. A 30-second delay is applied to account for the signal processing, diesel start, and damper closure time. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 115 cfm of unfiltered inleakage, and 375 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 97.5% for particulate, elemental iodine, and organic iodine.

The atmospheric dispersion factors (*X/Qs*) used for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3. Prior to the time of reactor trip, releases are from the condenser SJAE to the normal intake. Immediately following reactor trip, releases from the steam generators are assumed to occur from the MSSV/ADV which produces the most limiting *X/Q*. The receptor point shifts to the most liming emergency intake after control room isolation occurs.

The EAB and LPZ doses are determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are provided in Table 1.8.2-1.

2.4.5 Radiological Consequences

The radiological consequences of the SGTR Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Two activity release cases corresponding to the RCS maximum pre-accident iodine spike and the accident-induced iodine spike, based on proposed TS 3.4.8 limits, are analyzed. As shown in Table 2.4-6, the radiological consequences of the Turkey Point SGTR event for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.5 <u>Locked Rotor</u>

2.5.1 Background

This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system through the steam generator via the ADVs and MSSVs. In addition, radioactive iodine contained in the secondary inventory prior to the event is released to the atmosphere as a result of steaming from the steam generators following the accident. This event is described in Section 14.1.9 of the UFSAR.

2.5.2 Compliance with Reg. Guide 1.183 Regulatory Positions

The revised Locked Rotor dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on Reg. Guide 1.183, Regulatory Position 3.1, and is provided in Table 1.7.4-1. The inventory provided in Table 1.7.4-1 is then adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The fraction of fission product inventory in the gap available for release due to fuel breach is consistent with Table 3 of Reg. Guide 1.183.

2. Regulatory Position 2 - Fuel damage is assumed for this event.

3. Regulatory Position 3 - Activity released from the damaged fuel is assumed to mix instantaneously and homogeneously throughout the primary coolant.

4. Regulatory Position 4 - The chemical form of radioiodine released from the damaged fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to equilibrium iodine concentrations in the RCS and secondary system.

5. Regulatory Position 5.1 - The primary-to-secondary leak rate is equal to the value specified by proposed TS 6.8.4.j.b.2, which is 0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions.

6. Regulatory Position 5.2 - The density used in converting primary-to-secondary volumetric leak rates to mass leak rates is 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

7. Regulatory Position 5.3 - The release of radioactivity is assumed to continue until RHR is capable of removing decay heat and for providing for any further cooldown, which occurs at 63 hours.

8. Regulatory Position 5.4 - The analysis assumes a coincident loss of offsite power in the evaluation of fission products released from the secondary system.

9. Regulatory Position 5.5 - All noble gas radionuclides released from the primary system are assumed released to the environment without reduction or mitigation.

10. Regulatory Position 5.6 - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:

    - Appendix E, Regulatory Position 5.5.1 - A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncovery is postulated. The flashed leakage is assumed to be released to the environment with no mitigation. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

    - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the bulk water is credited.

- Appendix E, Regulatory Position 5.5.3 - All of the steam generator leakage flow that does not immediately flash is assumed to mix with the bulk water.

- Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the steam generator carryover rate of less than 1%.

- Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery in the unaffected steam generators is postulated for up to 30 minutes following a reactor trip. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.5.3 Other Assumptions

1. Reg. Guide 1.183, Section 3.6 - The assumed amount of fuel damage caused by the non-LOCA events is analyzed to determine the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and to determine the fraction of fuel elements for which fuel clad is breached. This analysis assumes DNB as the fuel damage criterion for estimating fuel damage for the purpose of establishing radioactivity releases. For the Locked Rotor event, Table 3 of Reg. Guide 1.183 specifies noble gas, alkali metal, and iodine fuel gap release fractions for the breached fuel.

2. This analysis assumes that the DNB fuel damage is limited to 15% breached fuel assemblies.

3. Radionuclide concentrations in the secondary side fluid of the steam generators assume that Auxiliary Feedwater is provided to maintain a constant secondary mass during periods of steam release.

4. The steam mass release rates are provided in Table 2.5-2.

5. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

2.5.4 Methodology

Input assumptions used in the analysis of the dose consequences of the Locked Rotor event are provided in Table 2.5-1. This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Following the reactor trip, the heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, the coolant temperatures will rise. The rapid rise in primary system temperatures during the initial phase of the transient results in a reduction in the initial DNB margin and fuel damage.

For the purpose of this dose assessment, a total of 15% of the fuel assemblies are assumed to experience DNB. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant. The source term is based upon release fractions from Table 3 of RG 1.183

which have been increased by the radial peaking factor of 1.65. The core source term used as the basis for this activity is presented in Table 1.7.4-1. The analysis assumes that activity is released as reactor coolant enters the steam generators due to primary-to-secondary leakage. All noble gases associated with this leakage are assumed to be released directly to the environment. All primary-to-secondary leakage is assumed to continue until the primary system is cooled to 212 °F at 125.4 hours. Activity is released to the atmosphere via steaming from the steam generator ADVs and MSSVs until RHR is capable of removing decay heat and for providing for any further cooldown, which occurs at 63 hours. The release of the initial iodine content of the steam generator secondary is also considered. The source term for this activity is presented in Table 1.7.3-1.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air.
- For the secondary release, the Control Room is isolated on a high radiation reading at the normal intake monitors. A 60 second delay is applied to account for the time to reach the setpoint, signal processing, and damper closure time. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 115 cfm of unfiltered inleakage, and 375 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 97.5% for particulate, elemental iodine, and organic iodine.

The atmospheric dispersion factors (*X/Qs*) used for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3. Releases from the steam generators are assumed to occur from the MSSV/ADV which produces the most limiting *X/Q*.

The EAB and LPZ doses are determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are provided in Table 1.8.2-1.

2.5.5 Radiological Consequences

The radiological consequences of the Locked Rotor event are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.5-3, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.6 Rod Cluster Control Assembly (RCCA) Ejection

2.6.1 Background

This event consists of the ejection of a single RCCA. This event is the same as the Rod Ejection event referred to in Reg. Guide 1.183. The RCCA Ejection results in a reactivity insertion that leads to a core power level increase and subsequent reactor trip. Two RCCA Ejection cases are considered. The first case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere. The second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system. This event is described in the UFSAR, Section 14.2.6.

2.6.2 Compliance with Reg. Guide 1.183 Regulatory Positions

The RCCA Ejection dose consequence analysis is consistent with the guidance provided in Reg. Guide 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on Reg. Guide 1.183, Regulatory Position 3.1, and is provided in Table 1.7.4-1. The inventory provided in Table 1.7.4-1 is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The release fractions provided in Reg. Guide 1.183 Table 3 are adjusted to comply with the specific Reg. Guide 1.183 Appendix H release requirements. For both the containment and secondary release cases, the activity available for release from the fuel gap for fuel that experiences DNB is assumed to be 10% of the noble gas and iodine inventory in the DNB fuel. For the containment release case for fuel that experiences fuel centerline melt (FCM), 100% of the noble gas and 25% of the iodine inventory in the melted fuel is assumed to be released to the containment. For the secondary release case for fuel that experiences FCM, 100% of the noble gas and 50% of the iodine inventory in the melted fuel is assumed to be released to the primary coolant.

2. Regulatory Position 2 - Fuel damage is assumed for this event.

3. Regulatory Position 3 - For the containment release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the containment atmosphere. For the secondary release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the primary coolant and be available for leakage to the secondary side of the steam generators.

4. Regulatory Position 4 - The chemical form of radioiodine released from the damaged fuel to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Containment sump pH is controlled to 7.0 or higher.

5. Regulatory Position 5 - The chemical form of radioiodine released from the steam generators to the environment is assumed to be 97% elemental iodine, and 3% organic iodide.

6. Regulatory Position 6.1 - For the containment leakage case, natural deposition in the containment is credited. Containment sprays are not credited in the mitigation of this event.

7. Regulatory Position 6.2 - The containment is assumed to leak at the proposed TS maximum allowable rate of 0.20% for the first 24 hours and 0.10% for the remainder of the event.

8. Regulatory Position 7.1 - The primary-to-secondary leak rate is equal to the value specified by proposed TS 6.8.4.j.b.2, which is 0.6 gpm through all steam generators and 0.2 gpm through any one steam generator at room temperature conditions.

9. Regulatory Position 7.2 -. The density used in converting primary-to-secondary volumetric leak rates to mass leak rates is 62.4 lbm/ft^3, which is consistent with the leakage limits at room temperature conditions.

10. Regulatory Position 7.3 - All of the noble gas released to the secondary side is assumed to be released directly to the environment without reduction or mitigation.

11. Regulatory Position 7.4 - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:

    - Appendix E, Regulatory Position 5.5.1 – A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncovery is postulated. The flashed leakage is assumed to be released to the environment with no mitigation. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.

- Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the bulk water is credited.

- Appendix E, Regulatory Position 5.5.3 - All of the steam generator leakage flow that does not immediately flash is assumed to mix with the bulk water.

- Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the steam generator carryover rate of less than 1%.

- Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery in the unaffected steam generators is postulated for up to 30 minutes following a reactor trip. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the steam generator into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator.

2.6.3 Other Assumptions

1. This analysis assumed that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS 3.7.1.4 limit of 0.1 μCi/gm Dose Equivalent I-131.

2. Radionuclide concentrations in the secondary side fluid of the steam generators assume that Auxiliary Feedwater is provided to maintain a constant secondary mass during periods of steam release.

3. The steam mass release rates for the secondary release are provided in Table 2.6-2.

4. It is assumed that 0.25% of the fuel is assumed to experience melting and 10% of the fuel is breached due to DNB.

2.6.4 Methodology

Input assumptions used in the dose consequence analysis of the RCCA Ejection are provided in Table 2.6-1. The postulated accident consists of two cases. One case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere, and the second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system. The RCCA Ejection is evaluated with the assumption that 0.25% of the fuel experiences FCM and 10% of the fuel experiences DNB. The release fractions from the damaged fuel correspond to the requirements set out in Regulatory Position 1 of Appendix H to Reg. Guide 1.183.

For the containment release case, 100% of the activity is released instantaneously to the containment. Natural deposition of the released activity inside of containment is credited. Radionuclide removal by the Emergency Containment Filters and containment spray is not credited. The containment is assumed to leak at the proposed TS maximum allowable rate of 0.20% for the first 24 hours and 0.10% for the remainder of the event.

For the secondary release case, primary coolant activity is released into the steam generators by leakage across the steam generator tubes. The core source term used as the basis for this activity is presented in Table 1.7.4-1. Core activities are then increased by the radial peaking factor of 1.65 for this event. All noble gases associated with this leakage are assumed to be released directly to the environment. Secondary activity is then released to the atmosphere via steaming from the MSSVs/ADVs until the RHR system is capable of removing decay heat and for providing for any further cooldown. The release of the initial iodine content of the steam generator secondary is also considered. The source term for this activity is presented in Table 1.7.3-1.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air.
- For the secondary release, the Control Room is isolated on a high radiation reading at the normal intake monitors. A 60 second delay is applied to account for the time to reach the setpoint, signal processing, and damper closure time. For the containment release, Control Room isolation occurs on high radiation on the containment radiation monitors. The 60 second delay time is conservatively applied to this release model. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 115 cfm of unfiltered inleakage, and 375 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 97.5% for particulate, elemental iodine, and organic iodine.

The atmospheric dispersion factors (*X/Qs*) used for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3. Releases from the steam generators are assumed to occur from the MSSV/ADV which produces the most limiting *X/Q*. Atmospheric dispersion factors for the containment release correspond to the nearest containment penetration.

The EAB and LPZ doses are determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are provided in Table 1.8.2-1.

The secondary release scenario credits control room isolation from a high radiation signal on the control room intake monitor. The Technical Specification setpoint for this instrument is 2 mR/hr. In the RCCA Ejection analysis, an analytical setpoint of 5 mR/hr was used to account for measurement and test uncertainties and to apply additional conservatism. For the design basis fuel failure and core melt fractions, the calculated exposure rate at the detector exceeded the analytical setpoint by approximately 35%. It was recognized that with only 35% margin, a scenario could be postulated with fuel failure fractions less than the design values in which the analytical setpoint would not be reached and a delayed manual isolation must be assumed. While the offsite dose consequences would be lower in such a scenario, the relative impact of lower fuel failure fractions with a longer control room isolation time was not immediately obvious. Therefore, an additional case was performed which combined the reduced source term with a 30-minute control room isolation time.

2.6.5 Radiological Consequences

The radiological consequences of the RCCA Ejection are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.6-3, the results of both cases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.7 Waste Gas Decay Tank (WGDT) Rupture

2.7.1 Background

This event involves a major rupture of one of the Waste Gas Decay Tanks as currently presented in Section 14.2.3 of the Turkey Point UFSAR. This analysis assumes that the ruptured WGDT contains an inventory equivalent to the equilibrium RCS noble gas activity from operation with 1% fuel defects. The leak rate from the WGDT to the environment simulates a major tank rupture which instantaneously releases the entire contents of the tank. No credit is taken for hold-up, dilution, or filtration in the Reactor Auxiliary Building.

2.7.2 Compliance with Reg. Guide 1.183 Regulatory Positions

Reg. Guide 1.183 does not provide direct guidance relative to the Waste Gas Decay Tank Rupture event. Guidelines for the review of WGDT analyses are given in Branch Technical Position 11-5 of the Standard Review Plan, with additional instruction available from Regulatory Issue Summery 2006-04. Compliance of the WGDT rupture analysis with the positions of BTP 11-5 is discussed below.

1. Position B.1.A – The analysis demonstrates that the exposure to an individual at the exclusion area boundary following a WGDT rupture does not exceed 0.1 rem TEDE. This limit is consistent with the guidance provided in Issue #11 of RIS 2006-04.

2. Position B.1.B - The analysis source term is equal to the RCS noble gas inventory resulting from extended full power operation with 1% defective fuel, and the entire RCS inventory is transferred to the WGDT following a shutdown.

   Exception: BTP 11-5 identifies the PWR-GALE code as an acceptable method for calculating the WGDT source term. However, the source term applied in the Turkey Point WGDT analysis is derived from ORIGEN 2.1 per the guidance of Section 3.1 of Reg. Guide 1.183. The WGDT source term is consistent with the RCS source term listed in Table 1.7.2-1 used in other applicable AST event analyses.

3. Position B.1.C - This analysis assumes an instantaneous failure of a batch-type waste gas system and a release through a pathway not normally used for planned releases. The release is from the building which houses the WGDT, and no effluent monitoring or isolation is assumed. The source term consists of only noble gases since particulates and iodines have been removed by other processes. The release is modeled as a ground level release without credit for a building wake factor. The EAB and LPZ X/Qs represent the maximum values from all downwind sectors and are conservative with respect to 5% overall site values. No deposition is assumed to occur downwind of the release.

   Exception: BTP 11-5 identifies an acceptable method for determining X/Qs outlined in SRP Section 2.3.4. However, the X/Qs used in the WGDT analysis were developed using the PAVAN code in accordance with Reg. Guide 1.145 per Section 5.3 of Reg. Guide 1.183. The release point and corresponding atmospheric dispersion factors are the same as those applied to releases from the Auxiliary Building in the analysis of the LOCA event in discussed in Section 2.1.

2.7.3 Other Assumptions

1. The WGDT source term provided in Table 2.7-2 is based upon the plant operating at a power level of 2652 MW_{th} with one percent failed fuel for an extended period of time sufficient to achieve equilibrium radioactive concentrations in the reactor coolant system. The entire noble gas inventory of the reactor coolant system is then assumed to be stripped and placed into a single WGDT. This inventory was calculated to be equal to 84,274.8 Curies Dose Equivalent Xe-133, which exceeds the Tech Spec LCO 3.7.9 limit of 70,000 Curies.

2. Since the control room recirculation filters do not remove noble gas isotopes, the sensitivity to control room isolation is primarily dependent upon the magnitude of the total flow rate into the control room and the relative size of the atmospheric dispersion factors. Since both of these parameters are larger when the control room ventilation system is in the normal alignment, it is assumed that the control room remains unisolated for the duration of this event.

2.7.4 Methodology

The input assumptions used in the dose consequence analysis of the WGDT are provided in Table 2.7-1. The dose assessment model releases the above-prescribed inventory from the tank at a high rate of release to simulate the tank rupture. The contents are released to the environment without any hold up, dilution or filtration.

The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room intake. The WGDT tank area is served by the Auxiliary Ventilation Exhaust Fans. However, since the exhaust fans are not required to be operable by the Technical Specifications, and since the Control Room is assumed to remain in the normal operating mode, the limiting *X/Qs* correspond a release from Auxiliary Building vent V-10 to the normal intake as shown in Tables 1.8.1-2 and 1.8.1-3. The EAB and LPZ doses are determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are provided in Table 1.8.2-1.

2.7.5 Radiological Consequences

Reg. Guide 1.183 does not provide specific requirements or dose limits for a WGDT failure. The TEDE limit of 0.1 rem at the EAB is established by Regulatory Issue Summary 2006-04. Since this limit is referenced to dose restrictions to the member of the public during normal operation from 10CFR Part 20, EAB dose will be evaluated over a 30-day period. Although no limit is given for the LPZ boundary, this location will also be evaluated against the EAB limit of 0.1 rem acceptance criteria. A TEDE limit 5.0 from 10CFR50.67 will be applied to the control room.

Area	Dose Criteria
EAB	0.1 rem TEDE (for 30 days)
LPZ	0.1 rem TEDE (for 30 days)
Control Room	5 rem TEDE (for 30 days)

As shown in Table 2.7-3, the radiological consequences of the Waste Gas Decay Tank Rupture are all within the appropriate acceptance criteria.

2.8 Spent Fuel Cask Drop

2.8.1 Background

This event considers the drop of a spent fuel transfer cask into the spent fuel pool as described in Section 14.2.1.3 of the Turkey Point UFSAR. Reg. Guide 1.183 does not provide any specific guidance for the Spent Fuel Cask Drop event; therefore, the requirements of the Fuel Handling Accident in Appendix B of the Reg. Guide are followed for the cask drop reanalysis.

Per Tech. Spec. 3.9.12, 1525 hours of decay is required before movement of the spent fuel cask is allowed. The analysis assumes that all 157 assemblies of a recently discharged core are assumed to be damaged by the cask drop. Thus, the core source term presented in Table 1.7.4-1 is used in the analysis. High burnup gap release fractions are also applied. This source term is allowed to decay for 1525 hours prior to release.

A minimum water level of 23 feet is maintained above the damaged fuel. This water level ensures an elemental iodine decontamination factor of 285 per the guidance provided in NRC Regulatory Issue Summary 2006-04.

2.8.2 Compliance with RG 1.183 Regulatory Positions

Regulatory Guide 1.183 does not provide specific guidance for the Spent Fuel Cask Drop event: However, the guidance for the Fuel Handling Accident in Appendix B of the Reg. Guide is judged to be closely applicable to the conditions of this event. Therefore the following discussion for refers to the Reg. guide 1.183 positions as stated in Appendix B for the Fuel Handling Accident:

1. Regulatory Position 1.1 - The amount of fuel damage is assumed to be equal to 157 assemblies.

2. Regulatory Position 1.2 - The fission product release from the breached fuel is based on Regulatory Positions 3.1 and 3.2 of Reg. Guide 1.183. The gap activity available for release is modified from that specified by Table 3 of Reg. Guide 1.183 to account for high burnup fuel as described in Section 1.7.6. These modified gap releases are applied to the core source term provided in Table 1.7.4-1. This activity is assumed to be released from the fuel assembly instantaneously.

3. Regulatory Position 1.3 - The chemical form of radioiodine released from the damaged fuel into the spent fuel pool is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The cesium iodide is assumed to completely dissociate in the spent fuel pool resulting in a final iodine distribution of 99.85% elemental iodine and 0.15% organic iodine.

4. Regulatory Position 2 - A minimum water depth of 23 feet is maintained above the damaged fuel assembly. Therefore, a decontamination factor of 285 is applied to the elemental iodine and a decontamination factor of 1 is applied to the organic iodine. As a result, the breakdown of the iodine species above the surface of the water is 57% elemental and 43% organic. Guidance for the use of 285 for the elemental iodine decontamination factor is provided in NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms."

5. Regulatory Position 3 - All of the noble gas released is assumed to exit the pool without mitigation. All of the non-iodine particulate nuclides are assumed to be retained by the pool water.

6. Regulatory Position 4.1 - The radioactive material released from the fuel pool is assumed to be released from the building to the environment over a 2-hour period.

7. Regulatory Position 4.2 - No credit is taken for filtration of the release.

8. Regulatory Position 4.3 - No credit is taken for dilution of the release.

9. Regulatory Position 5 - The event does not occur in the containment.

2.8.3 Other Assumptions

The dose acceptance criteria for the Spent Fuel Cask Drop are assumed to be the same as those for the Fuel Handling Accident.

2.8.4 Methodology

The input assumptions used in the dose consequence analysis of the Spent Fuel Cask Drop are provided in Table 2.8-1. 100% of the gap activity specified in Table 3 of RG 1.183 is assumed to be instantaneously released from 157 assemblies into the fuel pool and transported to the environment over 2 hours. A minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event. All of the noble gas released from the damaged fuel is assumed to escape from the pool. All of the non-iodine particulates released from the damaged fuel are assumed to be retained by the pool. The iodine released from the damaged fuel is assumed to be composed of 99.85% elemental and 0.15% organic.

The source term meets the requirements of Regulatory Position 1 of Appendix B to RG 1.183. Since a full core of 157 assemblies is assumed to be damaged by the cask drop, the source term for the Spent Fuel Cask Drop event is equal to the core source term listed in Table 1.7.4-1 which is decayed by 1525 hours.

For this event, the Control Room ventilation system cycles through two modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 1000 cfm of unfiltered fresh air.

- The Control Room is assumed to be manually isolated by the operators 30 minutes after the beginning of the event. After isolation, the air flow distribution consists of 525 cfm of filtered makeup flow from the outside, 115 cfm of unfiltered inleakage, and 375 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 97.5% for particulate, elemental iodine, and organic iodine.

The atmospheric dispersion factors (*X/Qs*) used for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Tables 1.8.1-2 and 1.8.1-3. The atmospheric dispersion factors applied to the Spent Fuel Cask Drop event are based upon a release from the Unit 4 spent fuel pool.

The EAB and LPZ doses are determined using the *X/Q* factors for the appropriate time intervals. These *X/Q* factors are provided in Table 1.8.2-1.

2.8.5 Radiological Consequences

The radiological consequences of the Spent Fuel Cask Drop are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.8-2, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.9 Other Dose Consequences

The Turkey Point UFSAR, Appendix 8A, discusses equipment EQ due to a radiation environment. Reg. Guide 1.183, Regulatory Position 6, allows the licensee to use either the AST or TID-14844 assumptions for performing the required EQ analyses until such time as a generic issue related to the effect of increased cesium releases on EQ doses is resolved. Analysis performed by Sandia National Labs in Reference 5.44 found that for equipment exposed to sump water, the integrated doses calculated with AST exceeded those calculated with TID-14844 only after 42 days for a typical PWR. These findings were cited in the closure of Generic Safety Issue 187 with no requirements to implement AST. Therefore, the Turkey Point EQ analyses will continue to be based on TID-14844 assumptions.

Positions 1.3 and 4.3 of Reg. Guide 1.183 identify various licensing requirements for which compliance may have been demonstrated, in part, by the evaluation of radiological consequences of design basis accidents. The Turkey Point responses to these licensing commitments are not being revised as a result of implementation of AST.

3.0 Summary of Results

Results of the Turkey Point radiological consequence analyses using the AST methodology and the corresponding allowable control room unfiltered inleakage are summarized on Table 3-1.

4.0 Conclusions

Full implementation of the Alternative Source Term methodology, as defined in Regulatory Guide 1.183, into the design basis accident analysis has been made to support control room habitability in the event of increases in control room unfiltered air inleakage. Analysis of the dose consequences of the Loss-of-Coolant Accident (LOCA), Fuel Handling Accident (FHA), Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Locked Rotor, Rod Cluster Control Assembly (RCCA) Ejection, Waste Gas Decay Tank (WGDT) Rupture, and Spent Fuel Cask Drop have been made using the Reg. Guide 1.183 methodology. The analyses used assumptions consistent with proposed changes in the Turkey Point licensing basis and the calculated doses do not exceed the defined acceptance criteria.

This report supports a maximum allowable control room unfiltered air inleakage of 115 cfm.

5.0 References

5.1 Turkey Point Units 3 and 4 Updated FSAR, Electronic Version, reviewed on 4/24/09.

5.2 TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 23, 1962.

5.3 USNRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000.

5.4 Code of Federal Regulations, 10CFR50.67, "Accident Source Term," revised 12/03/02.

5.5 NEI 99-03, "Control Room Habitability Guidance," Nuclear Energy Institute, Revision 0 dated June 2001 and Revision 1 dated March 2003.

5.6 NRC Generic Letter 2003-01, "Control Room Habitability," June 12, 2003.

5.7 Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

5.8 Federal Guidance Report No. 12 (FGR 12), "External Exposure to Radionuclides in Air, Water, and Soil," 1993.

5.9 Florida Power & Light Company Turkey Point Plant Units 3 and 4 Technical Specifications through Amendments 239.

5.10 Letter from G. Van Middlesworth to USNRC, "Duane Arnold Energy Center, Docket No: 50-331, Op. License No: DPR-49, Technical Specification Change Request (TSCR-037): "Alternative Source Term"," October 19, 2001.

5.11 Duane Arnold Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 240 to DPR-49 issued July 31, 2001.

5.12 ARCON96 Computer Code ("Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997, RSICC Computer Code Collection No. CCC-664 and July 1997 errata).

5.13 MicroShield Version 5 "User's Manual "and "Verification & Validation Report, Rev. 5," Grove Engineering, both dated October 1996.

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5.17 NAI 8907-06, Revision 16, GOTHIC Containment Analysis Package Technical Manual, Version 7.2a(QA), January 2006

5.18 NAI 8907-09, Revision 9, GOTHIC Containment Analysis Package Qualification Report, Version 7.2a(QA), January 2006

5.19 Numerical Applications Inc., NAI-9912-04, Revision 2, "RADTRAD-NAI Version 1.0p3(QA) Documentation," July 2002.

5.20 Numerical Applications Inc., "Dose Methodology Quality Assurance Procedures," Revision 1, June 4, 2001.

5.21 NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

5.22 USNRC, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, February 1983.

5.23 USNRC, Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

5.24 NUREG-0800, USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,".

5.25 Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983.

5.26 NUREG/CR-4102, "Air Currents Driven by Sprays in Reactor Containment Buildings," May 1986.

5.27 Fort Calhoun Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 198 to DPR-40 issued April 4, 2001.

5.28 USNRC, Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Rev. 3, June 2001.

5.29 NRC Information Notice 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," September 19, 1991.

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5.32 ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," approved September 21, 1999; including Errata dated December 1, 2005.

5.33 WCAP-16596-NP, "Evaluation of Alternative Emergency Core Cooling System Buffering Agents," Revision 0.

5.34 NUREG/CR-5009, Assessment of the Use of Extended Burnup Fuel in Light Water Reactors.

5.35 USNRC, Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors", March 1972.

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Changes; TSTF-51, TSTF-68, and TSTF-312, June 5, 2002. (ADAMS Accession No. ML021840136).

5.37 CP&L, Serial HNP-01-120, "Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Revision to Alternative Source Term Methodology Analyses in Support of the Steam Generator Replacement and Power Uprate License Amendment Applications", August 17, 2001.

5.38 NMC Letter NRC-02-024, "Revision to the Design Basis Radiological Analysis Accident Source Term", Docket 50-305, Operating License DPR-43, Kewaunee Nuclear Power Plant, March 19,2002.

5.39 Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 166 to Facility Operating License No. DPR-43, Nuclear Management Company LLC, Kewaunee Nuclear Power Plant, Docket No. 50-305. (ADAMS Accession No. ML0302100620).

5.40 Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 182 and 169 to Facility Operating License Nos. NPF-76 and NPF-80, STP Nuclear Operating Company, Et. Al., South Texas Project, Units 1 and 2, Docket Nos. 50-498 and 50-499. (ADAMS Accession No. ML072680192).

5.41 USNRC SER, Attachment to License Amendment 215, Facility Operating License DPR-64, Indian Point Nuclear Generating Unit 3 – Issuance of Amendment Re: Selective Adoption of Alternate Source Term (TAC No. MB5382), March 17, 2003. (ADAMS Accession No. ML030760135).

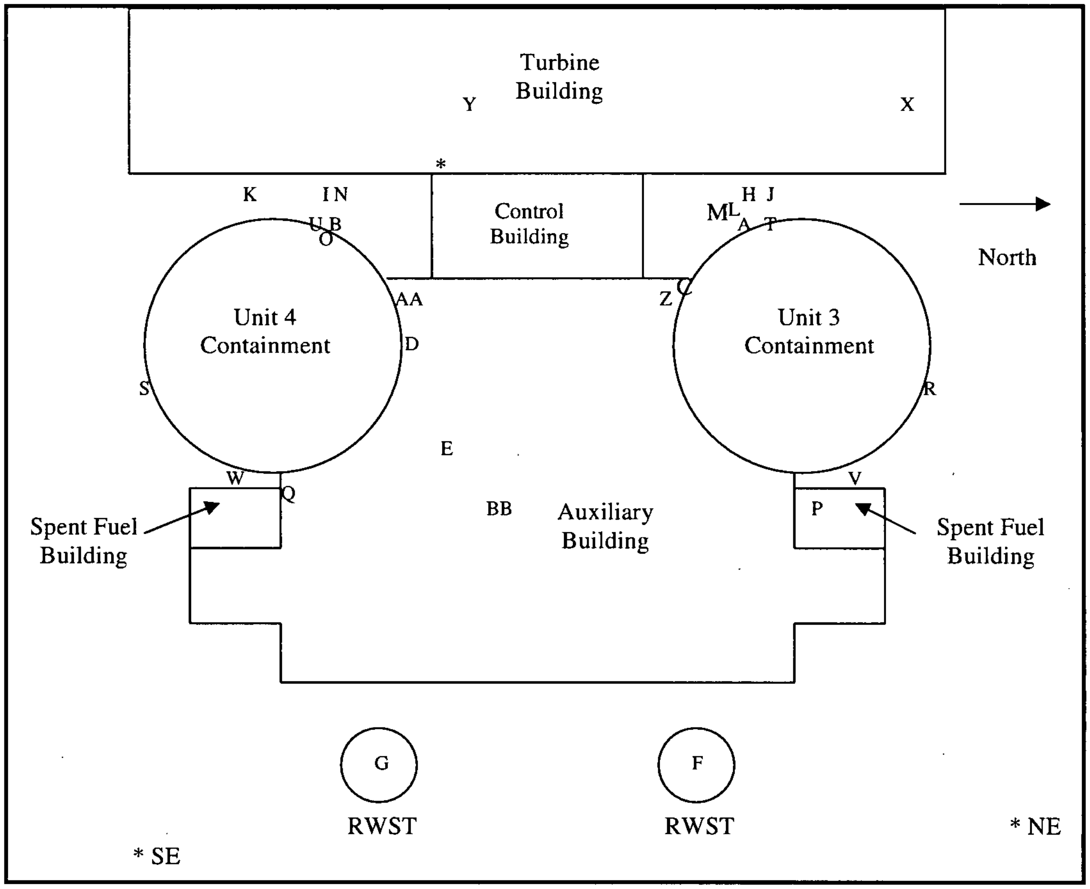
5.42 Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 226 to Renewed Facility Operating License No. DPR-20, Entergy Nuclear Operations, Inc, Palisades Plant, Docket No. 50-255, September 28 2007. (ADAMS Accession No. ML72470667).

5.43 Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 152 to Renewed Facility Operating License No. NPF-16, Florida Power and Light Company, St. Lucie Plant, Unit No. 2, September 29 2008, Docket No. 50-389. (ADAMS Accession No. ML082060400).

5.44 Evaluation of Radiological Consequences of Design Basis Accidents at Operating Reactors Using The Revised Source Term, Sandia National Laboratories, September 28, 1998.

Figure 1.8.1-1

Onsite Release-Receptor Location Sketch



(Not to scale)

A – Unit 3 Main Steam Line Closest Penetration
B – Unit 4 Main Steam Line Closest Penetration
C – Unit 3 Purge Duct Outlet
D – Unit 4 Purge Duct Outlet
E – Plant Stack
F – Unit 3 RWST
G – Unit 4 RWST
H – Unit 3 Closest MSSV
I – Unit 4 Closest MSSV
J – Unit 3 ADV Silencer
K – Unit 4 ADV Silencer
L – Unit 3 Main Stm Line Closest Point (Normal Intake)
M – Unit 3 Main Stm Line Closest Point (NE & SE Intakes)
N – Unit 4 Main Stm Line Closest Point (Normal Intake)
O – Unit 4 Main Stm Line Closest Point (NE & SE Intakes)
P – Unit 3 Spent Fuel Building
Q – Unit 4 Spent Fuel Building
R – Unit 3 Equipment Hatch
S – Unit 4 Equipment Hatch
T – Unit 3 Personnel Hatch
U – Unit 4 Personnel Hatch
V – Unit 3 Emergency Escape Lock
W – Unit 4 Emergency Escape Lock
X – Unit 3 SJAE
Y – Unit 4 SJAE
Z – Unit 3 Electrical Penetration
AA – Unit 4 Electrical Penetration
BB – Aux. Bldg Vent V-10
* – Normal Control Room Intake
* NE – Northeast Emergency CR Intake
* SE – Southeast Emergency CR Intake

Table 1.6.3-1
Control Room Ventilation System Parameters

Parameter	Value
Control Room Volume	47,786 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate and Inleakage	1000 cfm
Emergency Operation	
Recirculation Mode:	
Filtered Make-up Flow Rate	525 cfm
Filtered Recirculation Flow Rate	375 cfm
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage	115 cfm
Filter Efficiencies	
Elemental	97.5%
Organic	97.5%
Particulate	97.5%

Table 1.6.3-2
LOCA Direct Shine Dose

Source	Direct Shine Dose (rem)
Containment	0.059
Containment Purge Duct	0.333
CR Filters	0.270
External Cloud	0.061
Total	**0.723**

Table 1.7.2-1
Primary Coolant Source Term*

Nuclide	RCS Activity (µCi/g)	Nuclide	RCS Activity (µCi/g)
Co-58	3.507E-01	Xe-135	4.866E+00
Co-60	7.246E-02	Cs-134	1.023E+01
Kr-85	3.340E+01	Cs-136	5.439E+00
Kr-85m	1.138E+00	Cs-137	4.826E+00
Kr-87	6.856E-01	Ba-139	4.135E-04
Kr-88	2.024E+00	Ba-140	4.533E-03
Rb-86	8.063E-02	La-140	6.415E-03
Sr-89	3.216E-03	La-141	2.711E-04
Sr-90	2.956E-04	La-142	6.996E-05
Sr-91	1.237E-03	Ce-141	2.761E-02
Sr-92	5.302E-04	Ce-143	1.423E-03
Y-90	4.831E-04	Ce-144	6.536E-02
Y-91	3.077E-02	Pr-143	1.344E-02
Y-92	6.489E-04	Nd-147	4.428E-03
Y-93	4.249E-04	Kr-83m	2.931E-01
Zr-95	4.348E-02	Xe-131m	2.729E+00
Zr-97	7.967E-04	Xe-133m	3.138E+00
Nb-95	6.103E-02	Cs-138	6.838E-01
Mo-99	4.395E+00	Cs-134m	5.399E-02
Tc-99m	4.200E+00	Sb-124	4.464E-02
Ru-103	3.387E-02	Sb-125	7.716E-01
Ru-105	1.748E-04	Sb-126	8.069E-03
Ru-106	4.187E-02	Te-134	2.072E-02
Rh-105	1.413E-03	Te-125m	1.783E-01
Sb-127	2.078E-01	Te-133m	1.216E-02
Sb-129	2.819E-02	Rh-103m	3.379E-02
Te-127	5.701E-01	Nb-97	9.994E-05
Te-127m	3.888E-01	Nb-95m	3.216E-04
Te-129	3.988E-01	Pm-147	8.795E-03
Te-129m	5.687E-01	Pm-148	1.098E-03
Te-131m	8.316E-02	Pm-149	1.001E-03
Te-132	1.979E+00	Pm-151	1.844E-04
I-131	2.006E-01	Pm-148m	7.099E-04
I-132	1.422E-01	Y-91m	7.204E-04
I-133	2.431E-01	Br-82	2.147E-02
I-134	2.633E-02	Br-83	5.948E-02

Table 1.7.2-1
Primary Coolant Source Term*

Nuclide	RCS Activity (μCi/g)	Nuclide	RCS Activity (μCi/g)
I-135	1.200E-01	Br-84	2.554E-02
Xe-133	2.260E+02		

* The iodine activities have been adjusted to equal 0.25 μCi/gm DE I-131, and the non-iodine activities correspond to 447.7 μCi/gm DE Xe-133 based upon the proposed Technical Specification definition.

Table 1.7.3-1
Secondary Side Source Term

Nuclide	Activity (μCi/g)
I-131	8.022E-02
I-132	5.688E-02
I-133	9.725E-02
I-134	1.053E-02
I-135	4.801E-02

Table 1.7.4-1
Core Source Term

Nuclide	Containment Leakage Source (Curies)	Nuclide	Containment Leakage Source (Curies)
Co-58	0.00E+00	Pu-239	3.021E+04
Co-60	0.00E+00	Pu-240	5.103E+04
Kr-85	9.615E+05	Pu-241	1.287E+07
Kr-85m	1.813E+07	Am-241	1.403E+04
Kr-87	3.468E+07	Cm-242	5.267E+06
Kr-88	4.878E+07	Cm-244	8.359E+05
Rb-86	2.251E+05	I-130	4.475E+06
Sr-89	6.702E+07	Kr-83m	8.629E+06
Sr-90	7.655E+06	Xe-138	1.170E+08
Sr-91	8.252E+07	Xe-131m	8.291E+05
Sr-92	8.958E+07	Xe-133m	4.595E+06
Y-90	7.990E+06	Xe-135m	2.970E+07

Table 1.7.4-1
Core Source Term

Nuclide	Containment Leakage Source (Curies)	Nuclide	Containment Leakage Source (Curies)
Y-91	8.684E+07	Cs-138	1.299E+08
Y-92	8.999E+07	Cs-134m	5.657E+06
Y-93	1.043E+08	Rb-88	4.963E+07
Zr-95	1.192E+08	Rb-89	6.351E+07
Zr-97	1.173E+08	Sb-124	2.030E+05
Nb-95	1.205E+08	Sb-125	1.524E+06
Mo-99	1.355E+08	Sb-126	1.217E+05
Tc-99m	1.186E+08	Te-131	6.530E+07
Ru-103	1.270E+08	Te-133	8.486E+07
Ru-105	9.731E+07	Te-134	1.170E+08
Ru-106	5.914E+07	Te-125m	3.302E+05
Rh-105	8.808E+07	Te-133m	5.184E+07
Sb-127	9.265E+06	Ba-141	1.152E+08
Sb-129	2.581E+07	Ba-137m	9.693E+06
Te-127	9.194E+06	Pd-109	3.639E+07
Te-127m	1.237E+06	Rh-106	6.552E+07
Te-129	2.540E+07	Rh-103m	1.144E+08
Te-129m	3.785E+06	Tc-101	1.247E+08
Te-131m	1.110E+07	Eu-154	1.284E+06
Te-132	1.036E+08	Eu-155	8.731E+05
I-131	7.410E+07	Eu-156	2.598E+07
I-132	1.056E+08	La-143	1.066E+08
I-133	1.433E+08	Nb-97	1.183E+08
I-134	1.563E+08	Nb-95m	8.547E+05
I-135	1.346E+08	Pm-147	1.044E+07
Xe-133	1.437E+08	Pm-148	2.165E+07
Xe-135	3.779E+07	Pm-149	4.768E+07
Cs-134	2.184E+07	Pm-151	1.623E+07
Cs-136	6.015E+06	Pm-148m	2.573E+06
Cs-137	1.023E+07	Pr-144	9.855E+07
Ba-139	1.270E+08	Pr-144m	1.175E+06
Ba-140	1.231E+08	Sm-153	4.971E+07
La-140	1.278E+08	Y-94	1.053E+08
La-141	1.158E+08	Y-95	1.134E+08
La-142	1.118E+08	Y-91m	4.790E+07

Table 1.7.4-1
Core Source Term

Nuclide	Containment Leakage Source (Curies)	Nuclide	Containment Leakage Source (Curies)
Ce-141	1.179E+08	Br-82	6.087E+05
Ce-143	1.073E+08	Br-83	8.605E+06
Ce-144	9.786E+07	Br-84	1.487E+07
Pr-143	1.064E+08	Am-242	8.269E+06
Nd-147	4.650E+07	Np-238	5.195E+07
Np-239	1.897E+09	Pu-243	5.861E+07
Pu-238	4.201E+05		

Table 1.7.5-1
Fuel Handling Accident Source Term

Nuclide	Curies	Nuclide	Curies
Co-58	0.000E+00	Pu-239	3.175E+02
Co-60	0.000E+00	Pu-240	5.363E+02
Kr-85	1.010E+04	Pu-241	1.353E+05
Kr-85m	1.906E+05	Am-241	1.474E+02
Kr-87	3.645E+05	Cm-242	5.536E+04
Kr-88	5.127E+05	Cm-244	8.785E+03
Rb-86	2.366E+03	I-130	4.703E+04
Sr-89	7.044E+05	Kr-83m	9.068E+04
Sr-90	8.045E+04	Xe-138	1.230E+06
Sr-91	8.672E+05	Xe-131m	8.714E+03
Sr-92	9.415E+05	Xe-133m	4.830E+04
Y-90	8.397E+04	Xe-135m	3.122E+05
Y-91	9.126E+05	Cs-138	1.365E+06
Y-92	9.458E+05	Cs-134m	5.945E+04
Y-93	1.096E+06	Rb-88	5.216E+05
Zr-95	1.253E+06	Rb-89	6.674E+05
Zr-97	1.233E+06	Sb-124	2.133E+03
Nb-95	1.267E+06	Sb-125	1.602E+04
Mo-99	1.424E+06	Sb-126	1.279E+03
Tc-99m	1.247E+06	Te-131	6.862E+05
Ru-103	1.335E+06	Te-133	8.918E+05
Ru-105	1.023E+06	Te-134	1.229E+06

Table 1.7.5-1
Fuel Handling Accident Source Term

Nuclide	Curies	Nuclide	Curies
Ru-106	6.216E+05	Te-125m	3.470E+03
Rh-105	9.257E+05	Te-133m	5.448E+05
Sb-127	9.737E+04	Ba-141	1.211E+06
Sb-129	2.713E+05	Ba-137m	1.019E+05
Te-127	9.662E+04	Pd-109	3.825E+05
Te-127m	1.300E+04	Rh-106	6.885E+05
Te-129	2.670E+05	Rh-103m	1.202E+06
Te-129m	3.978E+04	Tc-101	1.311E+06
Te-131m	1.167E+05	Eu-154	1.349E+04
Te-132	1.088E+06	Eu-155	9.176E+03
I-131	7.788E+05	Eu-156	2.731E+05
I-132	1.110E+06	La-143	1.120E+06
I-133	1.506E+06	Nb-97	1.244E+06
I-134	1.642E+06	Nb-95m	8.983E+03
I-135	1.414E+06	Pm-147	1.097E+05
Xe-133	1.510E+06	Pm-148	2.275E+05
Xe-135	3.972E+05	Pm-149	5.011E+05
Cs-134	2.295E+05	Pm-151	1.706E+05
Cs-136	6.321E+04	Pm-148m	2.704E+04
Cs-137	1.075E+05	Pr-144	1.036E+06
Ba-139	1.335E+06	Pr-144m	1.235E+04
Ba-140	1.293E+06	Sm-153	5.224E+05
La-140	1.344E+06	Y-94	1.107E+06
La-141	1.217E+06	Y-95	1.192E+06
La-142	1.175E+06	Y-91m	5.034E+05
Ce-141	1.239E+06	Br-82	6.397E+03
Ce-143	1.128E+06	Br-83	9.044E+04
Ce-144	1.028E+06	Br-84	1.563E+05
Pr-143	1.118E+06	Am-242	8.691E+04
Nd-147	4.887E+05	Np-238	5.460E+05
Np-239	1.993E+07	Pu-243	6.159E+05
Pu-238	4.415E+03		

Table 1.8.1-1
Release-Receptor Combination Parameters for Analysis Events

Release-Receptor Pair	Release Location	Receptor Location	Release Height (m)	Receptor Height (m)	Distance (m)	Direction (deg)	Building Area (m^2)
A	Plant stack	Normal	55.5	4.3	46.3	95	0.01
B	Plant stack	SE emergency	55.5	1.2	85.4	323	0.01
C	Unit 4 RWST	Normal	15.2	4.3	92.9	97	0.01
D	Unit 4 RWST	SE emergency	15.2	1.2	60.3	354	0.01
E	Unit 4 Closest MSSV	Normal	18.6	4.3	17.0	158	0.01
F	Unit 4 Closest MSSV	SE emergency	18.4	1.2	98.4	291	0.01
G	Unit 4 Main Steam Line Closest Point	Normal	11.2	4.3	18.5	157	0.01
H	Unit 4 Main Steam Line Closest Point	SE emergency	11.2	1.2	92.7	294	1254
I	Unit 4 Personnel Hatch	Normal	3.3	4.3	23.1	148	1254
J	Unit 4 Emergency Escape Lock	SE	11.1	1.2	63.8	307	1254
K	Unit 4 Spent Fuel Building (NW corner)	Normal	4.3	4.3	57.3	118	0.01
L	Unit 4 Spent Fuel Building (SE corner)	SE emergency	1.2	1.2	43.0	319	0.01
M	Unit 4 SJAE	Normal	7.5	4.3	9.4	331	0.01
N	Unit 4 Westernmost Electrical Penetration	Normal	4.0	4.3	22.7	113	1254
O	Auxiliary Building Vent V-10	Normal	4.9	4.3	52.4	86	0.01

Table 1.8.1-2
Onsite Atmospheric Dispersion (*X/Q*) Factors for Analysis Events

This table summarizes the results for *X/Q* factors for the control room intakes for the various accident scenarios. Values are presented for the normal air intake prior to intake isolation and the least favorable emergency air intake after Control Room isolation. The same atmospheric dispersion factor is applied to both the makeup flow and unfiltered inleakage for each release-receptor pair. These values are not adjusted for Control Room Occupancy Factors. Note that the letters that indicate the release-receptor pairs do not necessarily correspond with the release identification letters on Figure 1.8.1-1.

Release-Receptor Pair	Release Point	Receptor Point	0-2 hour *X/Q*	2-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
A	Plant stack	Normal intake	1.85E-03				
B (1)	Plant stack	SE emergency intake	9.02E-04	7.50E-04	2.76E-04	2.22E-04	1.53E-04
C	Unit 4 RWST	Normal intake	9.72E-04				
D (1)	Unit 4 RWST	SE emergency intake	1.89E-03	1.49E-03	6.00E-04	4.67E-04	3.49E-04
E	Unit 4 Closest MSSV/ADV (2)	Normal intake	1.33E-02 (3)				
F (1)	Unit 4 Closest MSSV/ADV (2)	SE emergency intake	6.79E-04 (3)	4.44E-04	1.84E-04	1.36E-04	8.47E-05
G	Unit 4 Main Steam Line Closest Point	Normal intake	1.53E-02				
H (1)	Unit 4 Main Steam Line Closest Point	SE emergency intake	7.32E-04	4.61E-04	1.94E-04	1.45E-04	9.21E-05
I	Unit 4 Personnel Hatch	Normal intake	1.03E-02				
J (1)	Unit 4 Emergency Escape Lock	SE emergency intake	1.44E-03	1.03E-03	3.89E-04	3.29E-04	2.27E-04
K	Unit 4 Spent Fuel Building (NW corner)	Normal intake	2.31E-03				
L (1)	Unit 4 Spent Fuel Building (SE corner)	SE emergency intake	3.27E-03	2.66E-03	1.00E-03	8.15E-04	5.99E-04
M	Unit 4 SJAE	Normal intake	6.43E-02 (4)				
N	Unit 4 Westernmost Electrical Penetration	Normal intake	1.14E-02				
O	Auxiliary Building Vent V-10	Normal intake	2.80E-03				

Table 1.8.1-2 Notes:

(1) This receptor location qualifies for the dual intake credit allowed by Section 3.3.2.2 of Reg. Guide 1.194. This credit is not applied to the values shown in this table; however, these values are reduced by a factor of 2 when applied in the event analyses.

(2) The atmospheric dispersion factor corresponding to the limiting MSSV or ADV is used for each time period. No distinction is made between automatic steam relief from the MSSVs and controlled releases from the ADVs for radiological purposes.

(3) This release location meets the requirements for the plume rise credit described in Section 6 of Reg. Guide 1.194. The 0-2 hour values shown in this table are reduced by a factor of 5 when used in the applicable event analyses.

(4) The distance from the Unit 4 SJAE to the normal intake is 9.4 meters as shown in Table 1.8.1-1. Section 3.4 of Reg. Guide 1.194 that states ARCON96 should not be used to address situations with distances of less than about 10 m. Therefore, the value in this table was derived using a $1/r^2$ relationship referenced to an ARCON96-calculated value at 20 meters. The $1/r^2$ approach was demonstrated to calculate conservative atmospheric dispersion factors with respect to values determined directly from ARCON96 at the same distance. For example, the 10-meter X/Q value determined in this manner is 5.68E-02 sec/m^3 compared with the ARCON96 calculated value of 5.02E-02 sec/m^3, a difference of 11.6%. For shorter distances, this approach becomes more conservative. At 9.4 meters, the ARCON96 result is 5.57E-02 sec/m^3 , which is 13.4% less than the 6.43E-02 sec/m^3 value used in the analysis.

Table 1.8.1-3
Release-Receptor Point Pairs Assumed for Analysis Events (1)

Event	Prior to CR Isolation	During CR Recirculation
LOCA:		
- Containment Leakage	N	J
- ECCS Leakage	n/a	D
- RWST Backleakage	n/a	D
- Containment Purge	A	B
FHA		
- Containment Release	I	J
- FHB Release	K	L
Spent Fuel Cask Drop	K	L
MSLB:		
- Break Release	G	H
- MSSV/ADV Release	E	F
SGTR	M, E (2)	F
Locked Rotor	E	F
RCCA Ejection:		
- Containment Leakage	N	J
- Secondary Side Release	E	F
WGDT Rupture	O	n/a

(1) Letters correspond to Release-Receptor pairs listed in Table 1.8.1-2

(2) Prior to reactor trip, the release receptor pair is from the SJAE to the normal intake. The release point changes to the MSSV/ADVs immediately after reactor trip, and the receptor point shifts to the southeast emergency intake following control room isolation.

Table 1.8.2-1
Offsite Atmospheric Dispersion (*X/Q*) Factors for Analysis Events

Time Period	EAB *X/Q* (sec/m^3)	LPZ *X/Q* (sec/m^3)
0-2 hours	1.71E-04*	3.76E-05
0-8 hours	9.84E-05	1.64E-05
8-24 hours	7.47E-05	1.08E-05
1-4 days	4.11E-05	4.38E-06
4-30 days	1.74E-05	1.20E-06

* With the exception of the WGDT Rupture, only the 0-2 hour EAB X/Q is used in the event analyses

Table 1.9-1
Post-LOCA Sump pH Analysis Results

Parameter	Minimum Value	Maximum Value
NaTB Level in Large Baskets (ft)	2.187	2.77
NaTB Level in Small Baskets (ft)	1.917	2.5
Dissolved Mass of NaTB at Switchover (lbm)	4637	4687
Total NaTB Mass (lbm)	11061	14264
pH at Switchover	7.000	7.005
Long Term pH	7.241	7.396

Table 2.1-1
Loss of Coolant Accident (LOCA) – Inputs and Assumptions

Input/Assumption	Value
Release Inputs:	
Core Power Level	2652 MW_{th}
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 –5.0 w/o
Initial RCS Equilibrium Activity	0.25 µCi/gm DE I-131 and 447.7 µCi/gm DE Xe-133 (Table 1.7.2-1)
RCS Mass (maximum)	397,544 lbm
Core Fission Product Inventory	Table 1.7.4-1
Containment Leakage Rate 0 to 24 hours after 24 hours	 0.20% (by weight)/day 0.10% (by weight)/day
LOCA release phase timing and duration	Table 2.1-2
Core Inventory Release Fractions (gap release and early in-vessel damage phases)	Reg. Guide 1.183, Sections 3.1, 3.2, and Table 2
<u>ECCS Systems Leakage (from 15 minutes to 30 days)</u> Sump Volume (minimum) ECCS Leakage (2 times allowed value) Flashing Fraction Chemical form of the iodine released from the ECCS leakage No filtration or credit for building dilution, released directly to the environment	 239,000 gallons (31,949.5 ft^3) 4,650 cc/hr Calculated – 0.092 Used for dose determination – 0.10 97% elemental, 3% organic

Table 2.1-1
Loss of Coolant Accident (LOCA) – Inputs and Assumptions

Input/Assumption	Value
RWST Back-leakage (from 15 minutes to 30 days)	
Sump Volume (minimum)	239,000 gallons (31,949.5 $ft.^3$)
ECCS Leakage to RWST (2 times allowed value)	0.1 gph
Flashing Fraction	0 % based on temperature of fluid reaching the RWST. Elemental iodine is released into tank space based upon partition factor.
RWST liquid/vapor elemental iodine partition factor	41.18
Elemental Iodine fraction in RWST	Table 2.1-4
Initial RWST Liquid Inventory (minimum at time of recirculation)	60,000 gallons
Release from RWST Vapor Space	Table 2.1-5
Containment Purge Release (unfiltered)	7,000 cfm for 8 seconds
Removal Inputs:	
Containment Aerosol/Particulate Natural Deposition (only credited in unsprayed regions)	0.1/hour
Total Containment Volume	1,550,000 ft^3
Surface Area for Wall Deposition	537,903 ft^2
Containment Elemental Iodine Wall Deposition	5.58/hour
Containment Sprayed Region Volume	534,442 ft^3
Spray Fall Height	70 feet
Volumetric Spray Flow Rate	2.986 ft^3/sec
Containment Upper Unsprayed Region Volume	643,864 ft^3
Containment Lower Unsprayed Region Volume (below operating deck)	371,694 ft^3
Flowrate between Sprayed and Upper Unsprayed Volumes	990,000 cfm
Flowrate between Upper Unsprayed and Lower Unsprayed Volumes	375,000 cfm
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Aerosols Time to reach DF of 50	 20 hr^{-1} 2.305 hours 6.44 hr^{-1} (reduced to 0.644 at 3.061 hours) Greater than 3.06 hours

Table 2.1-1
Loss of Coolant Accident (LOCA) – Inputs and Assumptions

Input/Assumption	Value
Spray Initiation Time	63.8 seconds
Control Room Ventilation System Isolation Signal Time of CR Isolation Unfiltered Inleakage	Table 1.6.3-1 High Containment Radiation 30 seconds 115 cfm
Containment Purge Filtration	0 %
Transport Inputs:	
Containment Leakage Release	Nearest containment penetration to CR ventilation intakes
ECCS Leakage	RWST vent
RWST Backleakage	RWST vent
Containment Purge	Plant stack
Personnel Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Breathing Rates	Reg. Guide 1.183, Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	Reg. Guide 1.183, Section 4.2.6

Table 2.1-2 LOCA Release Phases *

Phase	Onset	Duration
Gap Release	30 seconds	0.5 hours
Early In-Vessel	0.5 hours	1.3 hours

* From Reg. Guide 1.183, Table 4

Table 2.1-3
LOCA Time Dependent RWST Total Iodine Concentration*

Time (hours)	RWST Iodine Concentration (gm-atom/liter)
0	0
0.25	0.000E+00
0.50	2.797E-11
1.0	8.384E-11
4.0	4.191E-10
8.0	8.660E-10
12.0	1.313E-09
24.0	2.654E-09
48.0	5.335E-09
72.0	8.016E-09
96.0	1.070E-08
100.0	1.114E-08
150.0	1.673E-08
200.0	2.231E-08
250.0	2.790E-08
300.0	3.348E-08
350.0	3.906E-08
400.0	4.464E-08
450.0	5.022E-08
500.0	5.580E-08
550.0	6.137E-08
600.0	6.695E-08
650.0	7.253E-08
700.0	7.810E-08
720.0	8.033E-08

*Includes radioactive and stable iodine isotopes

Table 2.1-4
LOCA Time Dependent RWST Elemental Iodine Fraction

Time (hr)	Elemental Iodine Fraction
0.00	0.000E+00
0.25	0.000E+00
0.50	3.698E-05
1.0	1.109E-04
4.0	5.536E-04
8.0	1.143E-03
12.0	1.730E-03
24.0	3.485E-03
48.0	6.958E-03
72.0	1.038E-02
96.0	1.376E-02
100.0	1.432E-02
150.0	2.119E-02
200.0	2.788E-02
250.0	3.439E-02
300.0	4.072E-02
350.0	4.690E-02
400.0	5.292E-02
450.0	5.879E-02
500.0	6.452E-02
550.0	7.012E-02
600.0	7.559E-02
650.0	8.093E-02
700.0	8.616E-02
720.0	8.822E-02

Table 2.1-5
Adjusted Release Rate from the RWST

Time (hours)	Adjusted Release Rate (cfm)
0.00	0
0.25	1.225E-07
12.0	8.437E-07
72.0	1.546E-06
100.0	3.603E-06
300.0	6.796E-06
500.0	9.977E-06
600.0	1.089E-05
700.0	1.148E-05
720.0	1.148E-05

Table 2.1-6
LOCA Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
LOCA	5.85	1.58	4.87
Acceptance Criteria	25	25	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

Table 2.2-1
Fuel Handling Accident (FHA) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level Before Shutdown	2652 MW_{th}
Discharged Fuel Assembly Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Radial Peaking Factor	1.65
Number of Fuel Assemblies Damaged	1
Release Fraction from Breached Fuel	See Section 1.7.6
Delay Before Spent Fuel Movement	72 hours
Release Duration	2 hours
FHA Source Term for a Single Assembly	Table 1.7.5-1
Water Level Above Damaged Fuel Assembly	23 feet minimum
Iodine Decontamination Factors	Elemental – 285 Organic – 1
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Time of CR Isolation – Containment Release Time of CR Isolation – FHB Release Unfiltered Inleakage	Table 1.6.3-1 30 seconds - High Containment Radiation 30 minutes from Manual Isolation 115 cfm
Breathing Rates	Reg. Guide 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	Reg. Guide 1.183 Section 4.2.6

Table 2.2-2
Fuel Handling Accident Dose Consequences

Case	EAB Dose[(1)] (rem TEDE)	LPZ Dose[(2)] (rem TEDE)	Control Room Dose[(2)] (rem TEDE)
FHA in Containment	0.91	0.20	1.32
FHA in Fuel Handling Building	0.91	0.20	3.80
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose

Table 2.3-1
Main Steam Line Break (MSLB) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2652 MW_{th}
Initial RCS Equilibrium Activity	0.25 μCi/gm DE I-131 and 447.7 μCi/gm DE Xe-133 (Table 1.7.2-1)
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131 (Table 1.7.3-1)
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Iodine Spike Appearance Rate	500 times
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate	0.2 gpm/SG
Time to establish shutdown cooling and terminate steam release	63 hours
Time for RCS to reach 212°F and terminate SG tube leakage	125.4 hours
RCS Mass (minimum)	366,086 lbm
SG Secondary Side Mass	Faulted SG - 131,516.5 lbm Intact SGs - 67,707 lbm per SG
Release from Faulted SG	Instantaneous
Steam Release from Intact SGs	Table 2.3-2
Time to re-cover Intact SG Tubes	30 minutes
Tube Uncovery Flashing Fraction	11%
Steam Generator Secondary Side Partition Coefficients	Faulted SG – none Intact SGs – 100
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Isolation Signal Time of CR Isolation Unfiltered Inleakage	Table 1.6.3-1 Safety Injection 41.5 seconds 115 cfm
Breathing Rates Offsite Control Room	 Reg. Guide 1.183, Section 4.1.3 Reg. Guide 1.183, Section 4.2.6
Control Room Occupancy Factors	Reg. Guide 1.183 Section 4.2.6

Table 2.3-2
Intact SGs Steam Release Rate*

Time (hours)	Intact SGs Steam Release Rate (lbm/min)
0.0	2622
2.0	2058
3.0	1931
4.0	1814
5.0	1694
8.0	1070
11.0	965
16.0	864
24.0	820
63.0	0.0

* Stored energy above RHR entry conditions is released between 2 and 8 hours

Table 2.3-3
60 μCi/gm DE I-131 Activities

Isotope	Activity (μCi/gm)
Iodine-131	48.1440
Iodine-132	34.1280
Iodine-133	58.3440
Iodine-134	6.3192
Iodine-135	28.8000

Table 2.3-4
Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Letdown Flow	132 gpm
Identified RCS Leakage	10 gpm
Unidentified RCS Leakage	1 gpm
RCS Mass	397,544 lbm
I-131 Decay Constant	6.000E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

Table 2.3-5
Concurrent (500 x) Iodine Spike Appearance Rate

Isotope	Appearance Rate (Ci/min)	8-hour Production (Ci)
Iodine-131	53.90	25870
Iodine-132	101.83	48881
Iodine-133	76.17	36564
Iodine-134	38.22	18343
Iodine-135	50.50	24241

Table 2.3-6
MSLB Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
MSLB pre-accident iodine spike	0.03	0.02	1.53
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
MSLB concurrent iodine spike	0.05	0.04	1.57
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day
(3) Reg. Guide 1.183, Table 6
(4) 10CFR50.67

Table 2.4-1
Steam Generator Tube Rupture (SGTR) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2652 MW_{th}
Initial RCS Equilibrium Activity	0.25 μCi/gm DE I-131 and 447.7 μCi/gm DE Xe-133 (Table 1.7.2-1)
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131 (Table 1.7.3-1)
Maximum pre-accident spike iodine concentration	60 μCi/gm DE I-131
Iodine Spike Appearance Rate	335 times
Duration of accident-initiated spike	8 hours
Integrated Break Flow and Steam Release	Table 2.4-2
Break Flow Flashing Fraction	Prior to Reactor Trip - 21% Following Reactor Trip – 11%
Time of Reactor Trip	291 seconds
Time to isolate ruptured SG	30 minutes
Steam Generator Tube Leakage Rate	0.2 gpm/SG
Time to establish shutdown cooling and terminate intact steam release	63 hours
Time for RCS to reach 212°F and terminate SG tube leakage	125.4 hours
RCS Mass (minimum)	366,086 lbm
SG Secondary Side Mass	67,707 lbm per SG
Time to re-cover Intact SG Tubes	30 minutes
Tube Uncovery Flashing Fraction	11%
Secondary Side Partition Coefficients	SG (Flashed tube flow) – none SG (Non-flashed tube flow) – 100 Condenser – 100
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Isolation Signal Time of CR Isolation Unfiltered Inleakage	Table 1.6.3-1 Safety Injection 321 seconds 115 cfm
Breathing Rates Offsite Control Room	 Reg. Guide 1.183, Section 4.1.3 Reg. Guide 1.183, Section 4.2.6
Control Room Occupancy Factor	Reg. Guide 1.183, Section 4.2.6

Table 2.4-2
SGTR Mass Flow Rates [1]

Time (hours)	Break Flow into Ruptured SG (lbm/min)	Steam Release from Ruptured SG (lbm/min)	Steam Release from Unaffected SGs[2] (lbm/min)
0 – 0.0808	6507	64,800	129,600
0.0808– 0.5	4161	3579	4033
0.5 – 2	0	0	4033
2 – 8	0	0	2833
8 – 24	0	0	1525
24 – 63	0	0	1270

[1] Flowrate is assumed to be constant within the time period
[2] Stored energy above RHR entry conditions is released between 2 and 8 hours

Table 2.4-3
60 μCi/gm D.E. I-131 Activities

Isotope	Activity (μCi/gm)
Iodine-131	48.1440
Iodine-132	34.1280
Iodine-133	58.3440
Iodine-134	6.3192
Iodine-135	28.8000

Table 2.4-4
Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Letdown Flow	132 gpm
Identified RCS Leakage	10 gpm
Unidentified RCS Leakage	1 gpm
RCS Mass	397,544 lbm
I-131 Decay Constant	6.000E-5 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013178 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

Table 2.4-5
Concurrent (335 x) Iodine Spike Appearance Rate

Isotope	Appearance Rate (Ci/min)	8-hour Production (Ci)
Iodine-131	36.11	17333
Iodine-132	68.23	32750
Iodine-133	51.04	24498
Iodine-134	25.60	12290
Iodine-135	33.84	16241

Table 2.4-6
SGTR Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
SGTR pre-accident iodine spike	0.82	0.18	2.85
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	0.28	0.07	1.10
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day
(3) Reg. Guide 1.183, Table 6
(4) 10CFR50.67

Table 2.5-1
Locked Rotor – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2652 MW_{th}
Core Fission Product Inventory	Table 1.7.4-1
Initial Secondary Side Equilibrium Iodine Activity	0.1 µCi/gm DE I-131 (Table 1.7.3-1)
Gap Release Fraction	Reg. Guide 1.183, Section 3.2, Table 3
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 –5.0 w/o
Radial Peaking Factor	1.65
Fuel Failure	15% DNB
RCS Mass (minimum)	366,086 lbm
Steam Generator Tube Leakage Rate	0.2 gpm/SG
Time to establish shutdown cooling and terminate steam release	63 hours
Time for RCS to reach 212°F and terminate SG tube leakage	125.4 hours
Time to re-cover SG Tubes	30 minutes
Tube Uncovery Flashing Fraction	11%
SG Secondary Side Mass	67,707 lbm per SG
Secondary Side Mass Releases to environment	Table 2.5-2
Secondary Side Partition Coefficients	SG (Flashed leakage) – none SG (Non-flashed leakage) – 100
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Isolation Signal Time of CR Isolation Unfiltered Inleakage	Table 1.6.3-1 High Radiation on Control Room Intake Monitors 60 seconds 115 cfm
Breathing Rates Offsite Onsite	 Reg. Guide 1.183, Section 4.1.3 Reg. Guide 1.183, Section 4.2.6
Control Room Occupancy Factor	Reg. Guide 1.183, Section 4.2.6

Table 2.5-2
Locked Rotor Steam Release*

Time (hours)	Intact SGs Steam Release Rate (lbm/min)
0.0	2598
2.0	2143
3.0	2016
4.0	1900
5.0	1779
8.0	2598
11.0	965
16.0	864
24.0	820
63.0	0.0

* Stored energy above RHR entry conditions is released between 2 and 8 hours

Table 2.5-3
Locked Rotor Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Locked Rotor	0.58	0.62	1.33
Acceptance Criteria	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) Reg. Guide 1.183, Table 6
(4) 10CFR50.67

Table 2.6-1
RCCA Ejection – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2652 MW_{th}
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 –5.0 w/o
Radial Peaking Factor	1.65
Percent of Core in DNB Design Basis Manual CR Isolation Case	 10% 6.22%
Percent of Core with Centerline Melt Design Basis Manual CR Isolation Case	 0.25% 0.16%
Gap Release Fraction	Reg. Guide 1.183, Appendix H, Position 1
Core Fission Product Inventory	Table 1.7.4-1
Initial Secondary Side Equilibrium Iodine Activity	0.1 µCi/gm DE I-131 (Table 1.7.3-1)
Release From DNB Fuel	Section 1 of Appendix H to Reg. Guide 1.183
Release From Fuel Centerline Melt Fuel	Section 1 of Appendix H to RG 1.183
Secondary Side Partition Coefficients	SG (Flashed leakage) – none SG (Non-flashed leakage) – 100
Steam Generator Tube Leakage Rate	0.2 gpm/SG
Time to establish shutdown cooling and terminate steam release	63 hours
Time to re-cover SG Tubes	30 minutes
Tube Uncovery Flashing Fraction	11%
RCS Mass (minimum)	366,086 lbm
SG Secondary Side Mass	67,707 lbm per SG
Chemical Form of Iodine Released to Containment	Particulate – 95% Elemental – 4.85% Organic – 0.15%
Chemical Form of Iodine Released from SGs	Particulate – 0% Elemental – 97 % Organic – 3%
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Time of CR Isolation – Containment Release Time of CR Isolation – Secondary (Automatic) Time of CR Isolation – Secondary (Manual) Unfiltered Inleakage	Table 1.6.3-1 30 seconds - High Containment Radiation 60 seconds - High Radiation on CR Intake Monitors 30 Minutes from Manual Isolation 115 cfm

Table 2.6-1
RCCA Ejection – Inputs and Assumptions

Input/Assumption	Value
Breathing Rates	Reg. Guide 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	Reg. Guide 1.183 Section 4.2.6
Containment Volume	1.55E+06 ft^3
Containment Leakage Rate 0 to 24 hours after 24 hours	 0.20% (by weight)/day 0.10% (by weight)/day
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 5.58 hr^{-1} Organic Iodine – None

Table 2.6-2
RCCA Ejection Steam Release*

Time (hours)	Intact SGs Steam Release Rate (lbm/min)
0.0	2598
2.0	2143
3.0	2016
4.0	1900
5.0	1779
8.0	2598
11.0	965
16.0	864
24.0	820
63.0	0.0

* Stored energy above RHR entry conditions is released between 2 and 8 hours

Table 2.6-3
RCCA Ejection Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
RCCA Ejection – Containment Release	0.88	0.40	2.47
RCCA Ejection – Secondary Release (Automatic CR Isolation)	0.61	0.57	1.19
RCCA Ejection – Secondary Release (Manual CR Isolation)	0.38	0.36	3.46
Acceptance Criteria	6.3 (3)	6.3 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) Reg. Guide 1.183, Table 6
(4) 10CFR50.67

Table 2.7-1
Waste Gas Decay Tank (WGDT) Rupture – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2652 MW_{th}
WGDT inventory	RCS equilibrium with 1% fuel defects (Table 2.7-2). Equal to 84,274.8 Curies Dose Equivalent Xe-133.
Tank volume	525 ft^3
Tank leak rate (arbitrarily high)	1E+06 cfm
Control Room Ventilation System Time of CR Isolation Unfiltered Inleakage Makeup Flow	Table 1.6.3-1 Not isolated 115 cfm 1000 cfm
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Breathing Rates Offsite Control Room	 Reg. Guide 1.183, Section 4.1.3 Reg. Guide 1.183, Section 4.2.6
CR Occupancy Factors	Reg. Guide 1.183, Section 4.2.6

Table 2.7-2
WGDT Source Term (1)

Isotope	Tank Inventory (Curies)
Kr-85m	214.22
Kr-85	6286.02
Kr-87	129.06
Kr-88	381.02
Xe-131m	513.74
Xe-133	42555.99
Xe-133m	590.55
Xe-135	916.04
Xe-135m	82.41
Xe-138	85.94

(1) Tank activity equals total RCS equilibrium noble gas activity with 1% fuel defects

Table 2.7-3
WGDT Rupture Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (1) (rem TEDE)	Control Room Dose (1) (rem TEDE)
WGDT	0.08	0.02	0.33
Acceptance Criteria	0.1 (3)	0.1 (3)	5 (2)

(1) Integrated 30-day dose
(2) 10CFR50.67
(3) SRP BTP 11-5

Table 2.8-1
Spent Fuel Cask Drop– Inputs and Assumptions

Input/Assumption	Value
Core Power Level Before Shutdown	2652 MW_{th}
Core Average Fuel Burnup	45,000 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Core Fission Product Inventory	Table 1.7.4-1
Number of Fuel Assemblies Damaged	157
Gap Release Fraction	FHA high burnup gap fractions from Section 1.7.6
Delay Before Spent Fuel Movement	1525 hours
Release Duration	2 hours
Water Level Above Damaged Fuel Assemblies	23 feet minimum
Iodine Decontamination Factors	Elemental – 285 Organic – 1
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Tables 1.8.1-2 and 1.8.1-3
Control Room Ventilation System Time of CR Isolation Unfiltered Inleakage	Table 1.6.3-1 30 minutes from Manual Isolation 115 cfm
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 2.8-2
Spent Fuel Cask Drop Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Spent Fuel Cask Drop	0.40	0.09	2.05
Acceptance Criteria	6.3(3)	6.3(3)	5(4)

(1) Worst 2-hour dose
(2) Integrated 30-day
(3) FHA Criteria from Reg. Guide 1.183, Table 6
(4) 10CFR50.67

Table 3-1

Turkey Point Units No. 3 and 4
Summary of AST Analysis Results

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
LOCA	5.85	1.58	4.87
MSLB Pre-accident Iodine Spike	0.03	0.02	1.53
SGTR Pre-accident Iodine Spike	0.82	0.18	2.85
Acceptance Criteria	**≤ 25**(3)	**≤ 25**(3)	**≤ 5**(4)
MSLB Concurrent Iodine Spike	0.05	0.04	1.57
SGTR Concurrent Iodine Spike	0.28	0.07	1.10
Locked Rotor	0.58	0.62	1.33
Acceptance Criteria	**≤ 2.5** (3)	**≤ 2.5** (3)	**≤ 5** (4)
FHA – Containment Release	0.91	0.20	1.32
FHA – Fuel Building Release	0.91	0.20	3.80
Spent Fuel Cask Drop	0.40	0.09	2.05
RCCA Ejection – Containment	0.88	0.40	2.47
RCCA Ejection – Secondary (Automatic CR Isolation)	0.61	0.57	1.19
RCCA Ejection – Secondary (Manual CR Isolation)	0.38	0.36	3.46
Acceptance Criteria	**≤ 6.3**(3)	**≤ 6.3**(3)	**≤ 5**(4)
WGDT	0.08(2)	0.02	0.33
Acceptance Criteria	**≤ 0.1**(5)	**≤ 0.1**(5)	**≤ 5**(4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) Reg. Guide 1.183, Table 6
(4) 10CFR50.67
(5) SRP BTP 11-5

LICENSE AMENDMENT REQUEST
LAR 196

ALTERNATIVE SOURCE TERM
AND CONFORMING AMENDMENT

ATTACHMENT 4

LIST OF COMMITMENTS

Regulatory Commitments

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

REGULATORY COMMITMENTS	DUE DATE/EVENT
FPL will relocate the Control Room Ventilation System emergency air intakes prior to implementation of Alternative Source Term.	
FPL will install stainless steel wire mesh baskets to maintain pH during the sump recirculation phase following a Design Basis LOCA.	
FPL will supersede the commitment from PTN Letters L-2009-062 and L-2009-063 related to NaTB sampling with the proposed Technical Specification requirements for the Recirculation pH Control System.	
FPL will replace the aluminum fins on the normal containment coolers with copper fins.	
FPL will update the necessary procedures to implement the manual operator action for initiation of the emergency control room ventilation system.	

LICENSE AMENDMENT REQUEST

ALTERNATIVE SOURCE TERM AND CONFORMING AMENDMENT LAR 196

ATTACHMENT 5

INPUT/ASSUMPTIONS COMPARISON TABLES

TABLE 5-1
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS

Release Inputs

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Core Power Level	2346 MWt	2652 MWt	AST core power models planned EPU.
Core Average Fuel Burnup	40,000 MWD/MTU	45,000 MWD/MTU	AST core average burnup conservatively increased to accommodate a planned EPU.
Fuel Enrichment	3.0 – 4.5% U-235	3.0 - 5.0% U-235	AST maximum fuel enrichment conservatively increased to accommodate a planned EPU.
Initial RCS Equilibrium Activity	1.0 μCi/gm DEI-131 100/Ē μCi/gram of gross radioactivity	0.25 μCi/gm DEI-131 and 447.7 μCi/gm DEXe-133	AST values are consistent with proposed changes to more restrictive TS limits.
Maximum RCS Mass	366,086 lbm	397,544 lbm	Current – Nominal RCS no-load conditions AST – Maximum RCS solid conditions Since the initial RCS activity is assumed to be at TS limits, this maximizes the release to containment used in the calculation of the containment purge dose.

TABLE 5-1 (continued)
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Core Fission Product Inventory		Attachment 3 Table 1.7.4-1	AST source term conservatively bounds the planned EPU and current conditions.
I-131	6.17E7		
I-132	8.19E7		
I-133	1.26E8		
I-134	1.39E8		
I-135	1.18E8		
Kr-85m	1.64E7		
Kr-85	7.80E5		
Kr-87	3.15E7		
Kr-99	4.44E7		
Xe-131m	6.87E5		
Xe-133m	3.93E6		
Xe-133	1.26E8		
Xe-135m	2.46E7		
Xe-135	3.62E7		
Xe-138	1.05E8		

TABLE 5-1 (continued)
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Containment Leakage Rate			
0 to 24 hours	0.25% (by weight)/day	0.20% (by weight)/day	AST values are consistent with proposed changes to more restrictive TS limits.
After 24 hours	0.125% (by weight)/day	0.10% (by weight)/day	
LOCA Release Phase Timing and Duration			
Gap Release			
Onset	Instantaneous	30 seconds	
Duration	Instantaneous	30 minutes	
Early In-Vessel			
Onset	Instantaneous	30 minutes	
Duration	Instantaneous	1.3 hours	
Core Inventory Release Fractions (gap release and early in-vessel phases)	N/A	RG 1.183, Sections 3.1 and 3.2 and Table 2	Current – Instantaneous release of 50% of core iodine and 100% of noble gases.
ECCS Leakage			
Sump Volume (minimum)	N/A	239,000 gallons (31, 949.5 ft^3)	Current – Dose consequences from ECCS leakage not part of the current licensing basis. AST – Minimum sump volume maximizes sump activity concentration.
ECCS Leakage Duration	N/A	15 minutes to 30 days	

TABLE 5-1 (continued)
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
	ECCS Leakage (2 times allowable value)	N/A	4,650 cc/hr (2 times allowable value)	AST – The maximum allowable ECCS leakage value is doubled in accordance with RG 1.183.
	Flashing Fraction	N/A	9.2%	
	Airborne Iodine Fraction	N/A	10%	
Chemical Form of Iodine Released from ECCS Leakage				
	Elemental	N/A	97%	
	Organic	N/A	3%	
Initial RWST Liquid Inventory (minimum at time of recirculation		60,000 gallons	60,000 gallons	
Release from RWST Vapor Space		N/A	Attachment 3 Table 2.1-5	
Containment Purge Release (unfiltered)		7,000 cfm for 8 seconds	7,000 cfm for 8 seconds	
RWST Back-leakage				
	Sump Volume (minimum)	N/A	239,000 gallons (31, 949.5 ft^3)	Current – Dose consequences from RWST back-leakage are not part of the current licensing basis.
	RWST Back Leakage Duration	N/A	15 minutes to 30 days	
	ECCS Leakage	N/A	0.1 gph	
	Flashing Fraction	N/A	0% based on temperature of fluid reaching RWST.	
	RWST Liquid/Vapor Elemental Iodine Partition Factor	N/A	41.18	
	Elemental Iodine Fraction in RWST	N/A	Attachment 3 Table 2.1-4	

**TABLE 5-1 (continued)
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS**

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Removal Inputs				
Total Containment Volume		1,550,000 ft^3	1,550,000 ft^3	
Containment Sprayed Region Volume		Not credited	534,442 ft^3	Current – Containment spray is not credited in the current licensing dose consequences analysis.
Containment Upper Unsprayed Region Volume		Not credited	643,864 ft^3	
Containment Lower Unsprayed Region Volume		Not credited	371,694 ft^3	
Containment Natural Deposition Coefficients				
	Instantaneous Iodine Plateout at t=0	50%	0%	
	Aerosols	None	0.1 hr^{-1}	
	Elemental Iodine	5.94/hour for DF $\leq$ 100 0/hour for DF > 100	5.58 hr^{-1}	
	Organic Iodine	None	None	
Surface Area for Natural Deposition		569,219 ft^2	537,903 ft^2	AST – Total containment surface area participating in surface deposition was reduced by 25% for natural deposition of elemental iodine.

TABLE 5-1 (continued)
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Spray Fall Height		Not credited	70 feet	Current – Containment spray is not credited in the current licensing dose consequences analysis.
Volumetric Spray Flow Rate		Not credited	2.986 ft^3/sec	
Flowrate between Sprayed and Upper Unsprayed Volumes		Not credited	990,000 cfm	
Flowrate between Upper/Lower Unsprayed Volumes		Not credited	375,000 cfm	
Spray Removal Rates				
	Elemental Iodine	Not credited	20/hour	
	Time to reach DF of 200	Not credited	2.305 hours	
	Aerosol	Not credited	6.44/hour (reduced to 0.644 at 3.053 hours)	
	Time to reach DF of 50	Not credited	3.053 hours	
Spray Initiation Time		Not credited	63.8 seconds	

TABLE 5-1 (continued)
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Control Room Ventilation System			
Control Room Volume	47,786 ft^3	47,786 ft^3	
Normal Operation			
Filtered Makeup Flow Rate	0 cfm	0 cfm	
Filtered Recirculation Flow Rate	0 cfm	0 cfm	
Unfiltered Makeup Flow Rate and Inleakage	1,000 cfm	1,000 cfm	
Recirculation Mode			
Filtered Makeup Flow Rate	525 cfm	525 cfm	
Filtered Recirculation Flow Rate	375 cfm	375 cfm	
Unfiltered Makeup Flow Rate	0 cfm	0 cfm	
Unfiltered Inleakage	10 cfm	115 cfm	
Filter Efficiencies			AST - Filter efficiencies are consistent with proposed change to more restrictive TS.
Elemental	95%	97.5%	
Organic	95%	97.5%	
Particulate	95%	97.5%	
Time of Control Room Isolation	Control room isolated prior to arrival of plume	30 seconds	
Control Room Unfiltered Inleakage	10 cfm	115 cfm	

TABLE 5-1 (continued)
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Emergency Containment Filter Efficiency	2 ECFs credited	Not credited	
Particulate	95%		
Elemental	90%		
Methyl	30%		
Containment Purge Filtration	0%	0%	
Transport Inputs			
Containment Leakage Release	Containment to control room	Nearest containment penetration to CR ventilation intakes	
ECCS Leakage	N/A	Plant stack	Current – Dose consequences from ECCS leakage not part of the current licensing basis.
RWST Backleakage	N/A	RWST vent	Current – Dose consequences form RWST back-leakage are not part of the current licensing basis.
Containment Purge	Containment to control room	Plant stack	

TABLE 5-1 (continued)
LOSS OF COOLANT ACCIDENT (LOCA)
AST INPUTS AND ASSUMPTIONS

Personnel Dose Conversion Inputs

INPUT/ASSUMPTION			CURRENT VALUE	AST VALUE	REMARKS
Atmospheric Dispersion Factors Offsite			1.54 E-4 sec/m^3	Attachment 3 Table 1.8.2-1	AST - All Dispersion Factors recalculated using guidance from RG 1.194 and RG 1.145.
	Low Population Zone				
		0 – 2 hours	1.50 E-5 sec/m^3		
		2 – 12 hours	6.50 E-6 sec/m^3		
		12 -720 hours	2.40 E-7 sec/m^3		
	Control Room			Attachment 3 Tables 1.8.1-2 and 1.8.1-3	
		0 – 8 hours	9.58 E-4 sec/m^3		
		8 – 24 hours	7.52 E-4 sec/m^3		
		24 – 96 hours	5.26 E-4 sec/m^3		
		96 - 720 hours	2.94 E-4 sec/m^3		
Breathing Rates					
	Offsite 0-8 hours		3.47 E-4 m^3/sec	3.5 E-4 m^3/sec	
	8 – 24 hours		1.75 E-4 m^3/sec	1.8 E-4 m^3/sec	
	24 - 720 hours		2.32 E-4 m^3/sec	2.3 E-4 m^3/sec	
	Control Room Duration of Event		Same as Offsite	3.5 E-4 m^3/sec	

TABLE 5-1 (continued) LOSS OF COOLANT ACCIDENT (LOCA) AST INPUTS AND ASSUMPTIONS				
INPUT/ASSUMPTION		**CURRENT VALUE**	**AST VALUE**	**REMARKS**
Control Room Occupancy Factor				
	0 - 24 hours	1.0	100%	
	24 - 96 hours	0.6	60%	
	96 – 720 hours	0.4	40%	

TABLE 5-2
FUEL HANDLING ACCIDENT (FHA)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Core Power Level Before Shutdown	2346 MWt	2652 MWt	AST core power models planned EPU.
Discharged Fuel Assembly Burnup			
Core Average Burnup	40,000 MWD/MTU	45,000 MWD/MTU	AST core average burnup conservatively increased to accommodate a planned EPU.
Peak Average Burnup	62,000 MWD/MTU	62,000 MWD/MTU	Current & AST – gap release fraction for breached fuel increased to accommodate high burnup fuel; see Enclosure 3, Section 1.7.6.
Fuel Enrichment	3.0 – 4.5 % U-235	3.0 – 5.0 % U-235	AST maximum fuel enrichment conservatively increased to accommodate a planned EPU.
Core Average Assembly	4.5% U-235		
Peak Assembly	4.5% U-235		
Radial Peaking Factor	1.7	1.65	AST assumed radial peaking factor combined with the planned EPU core power value bounds the current radial peaking factor combined with the current power level.
Number of Fuel Assemblies Damaged	1	1	
Gap Release Fraction for Breached Fuel	0.10, except 0.12 (I-131) & 0.30 (Kr-85)	0.10, except 0.12 (I-131) & 0.30 (Kr-85)	Current & AST – gap release fraction for breached fuel increased to accommodate high burnup fuel; see Enclosure 3, Section 1.7.6.
Delay Before Spent Fuel Movement	72 hours	72 hours	
Release Duration	2 hours	2 hours	

TABLE 5-2 (continued)
FUEL HANDLING ACCIDENT (FHA)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
FHA Source Term for Single Assembly	UFSAR Table 14.2.1-1	Attachment 3 Table 1.7.5-1	AST source term conservatively bounds the planned EPU and current conditions.
Water Level Above Damaged Fuel Assembly	23 feet minimum	23 feet minimum	
Overall Iodine Decontamination Factor	200	200	Current & AST - An overall iodine contamination factor of 200 is used per RG 1.183, Appendix B. This is equivalent to a decontamination factor of 285 for elemental iodine and 1 for organic iodine per the guidance of RIS 2006-04.
Noble Gas Decontamination Factor	1	1	
Chemical Form of Iodine in Pool			
Elemental	57%	57%	Current & AST – pool chemical composition based on RG 1.183, Appendix B.
Organic	43%	43%	
Atmospheric Dispersion Factors			AST - All Dispersion Factors recalculated using guidance from RG 1.194 and RG 1.145.
Offsite (EAB)	1.54 E-04 sec/m^3	Attachment 3 Table 1.8.2-1	
Offsite (LPZ)	NA		
Onsite (Control Room)			
0 – 2 hours (FHA Puff Release)	1.51 E-03 sec/m^3	Attachment 3 Tables 1.8.1-2 and 1.8.1-3	
0 - 8 hours	9.58 E-04 sec/m^3		
8 – 24 hours	7.52 E-04 sec/m^3		
24 – 96 hours	5.26 E-04 sec/m^3		
96 – 720 hours	2.94 E-04 sec/m^3		

TABLE 5-2 (continued)
FUEL HANDLING ACCIDENT (FHA)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Control Room Ventilation System			
Control Room Volume	47,786 ft^3	47,786 ft^3	AST filter efficiency values are consistent with proposed changes to more restrictive TS limits.
Normal Operation			
Filtered Makeup Flow Rate	0 cfm	0 cfm	
Filtered Recirculation Flow Rate	0 cfm	0 cfm	
Unfiltered Makeup Flow Rate and Inleakage	1,000 cfm	1,000 cfm	
Recirculation Mode			
Filtered Makeup Flow Rate	525 cfm	525 cfm	
Filtered Recirculation Flow Rate	375 cfm	375 cfm	
Unfiltered Makeup Flow Rate	0 cfm	0 cfm	
Unfiltered Inleakage	500 cfm	115 cfm	
Filter Efficiencies			
Elemental	95%	97.5%	
Organic	95%	97.5%	
Particulate	N/A	97.5%	
Time of Control Room Isolation – Containment Release	30 seconds	30 seconds	Current – Dose consequences of the FHB release not specifically modeled and assumed equivalent to the containment release dose consequences. AST - FHB release specifically modeled in to address the recalculated Atmospheric Dispersion factors.

TABLE 5-2 (continued)
FUEL HANDLING ACCIDENT (FHA)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Breathing Rates			
Offsite (LPZ) 0 – 8 hours	3.47 E-04 m^3/sec	3.5 E-4 m^3/sec	
8 – 24 hours		1.8 E-4 m^3/sec	
After 24 hours		2.3 E-4 m^3/sec	
Offsite (EAB)	3.47 E-04 m^3/sec	1.71 E-4 m^3/sec	
Control Room Duration of the event	3.47 E-04 m^3/sec	3.5 E-4 m^3/sec	
Control Room Occupancy Factor			
0 – 24 hours	100%	100%	
24 – 96 hours	60%	60%	
96 – 720 hours	40%	40%	

TABLE 5-3
MAIN STEAM LINE BREAK (MSLB)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Core Power Level	2346 MWt	2652 MWt	AST core power models planned EPU.
Initial RCS Equilibrium Activity	1.0 µCi/gm DEI-131	0.25 µCi/gm DEI-131 and 447.7 µCi/gm DEXe-133	AST values are consistent with proposed changes to more restrictive TS limits.
Initial Secondary Side Equilibrium Iodine Activity	0.1 µCi/gm DEI-131	0.1 µCi/gm DEI-131	
Maximum Pre-Accident Spike Iodine Concentration	60 µCi/gm DEI-131	60 µCi/gm DEI-131	
Iodine Spike Appearance Rate	500 times	500 times	
Duration of Accident Initiated Spike	0.6 hours	8 hours	AST – The spike duration is consistent with the guidance in RG 1.183, Appendix E.
Steam Generator Tube Leakage Rate	500 gpd/SG at accident conditions	0.2 gpm/SG at room temperature conditions	AST – Accident leakage consistent with analysis assumptions and proposed change to SG performance criterion TS.
Time to Establish Shutdown Cooling and Terminate Steam Release	24 hours	63 hours	AST - Cooldown times conservatively bound the current and planned EPU conditions.
Time for RCS to Reach 212°F and Terminate SG Tube Leakage	24 hours	125.4 hours	
Minimum RCS Mass	366,086 lbm	366,086 lbm	Current - Minimum RCS mass bounds the current and EPU conditions due to higher steam generator tube plugging assumption (20% vs. 10%).

TABLE 5-3 (continued)
MAIN STEAM LINE BREAK (MSLB)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Steam Generator Secondary Side Mass			Current - Values are based on nominal mass at full power. AST – Conservatively assumes maximum SG mass for the faulted SG to maximize the pre-existing secondary activity release to the environment– and the minimum mass for the intact SGs to maximize the release due to RCS iodine spiking and bounds current and planned EPU conditions..
Faulted Steam Generator (Maximum)	84,128 lbm	131,516.5 lbm	
Intact Steam Generators (Minimum)	84,128 lbm per SG	67,707 lbm per SG	
Release from Faulted SG			AST - Steam releases conservatively bound current and planned EPU conditions
0 – 2 hours	84,128 lbm	131,516,5 lbm; Instantaneous	
Steam Release from Intact SGs			
0 – 2 hours	269,700 lbm	Attachment 3 Table 2.3-2	
2 – 8 hours	369,300 lbm		
8 – 24 hours	984,700 lbm		

TABLE 5-3 (continued)
MAIN STEAM LINE BREAK (MSLB)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Time to Re-Cover Intact SG Tubes	N/A	30 minutes	Current – Dose consequences from tube uncovery are not part of the current licensing basis.
Tube Uncovery Flashing Fraction	N/A	11%	
Steam Generator Secondary Side Partition Coefficients			Equivalent values used for both Current & AST Current – values are multipliers AST – value is a divisor
Faulted Steam Generator	1.0	None	
Intact Steam Generator	0.01	100	
Atmospheric Dispersion Factors			AST – All Dispersion Factors recalculated using guidance from RG 1.194 and RG 1.145.
Offsite		Attachment 3 Table 1.8.2-1	
0 – 2 hours (EAB)	1.54E-4 m^3/sec		
0 – 2 hours (LPZ)	1.5 E-5 m^3/sec		
2 – 12 hours (LPZ)	6.5 E-6 m^3/sec		
>12 hours (LPZ)	2.4 E-7 m^3/sec		
Onsite	N/A	Attachment 3 Tables 1.8.1-2 and 1.8.1-3	
Control Room Ventilation System	N/A	Attachment 3 Table 1.6.3-1	Current – CR doses not calculated – bounded by LOCA analysis.
Time of Control Room Isolation	N/A	41.5 seconds	
Control Room Unfiltered Inleakage	N/A	115 cfm	

TABLE 5-3 (continued)
MAIN STEAM LINE BREAK (MSLB)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Breathing Rates			
Offsite 0 – 8 hours	3.47 E-4 m^3/sec	3.5 E-4 m^3/sec	
8 – 24 hours	1.75 E-4 m^3/sec	1.8 E-4 m^3/sec	
After 24 hours	N/A	2.3 E-4 m^3/sec	
Control Room Duration of the event	N/A	3.5 E-4 m^3/sec	
Control Room Occupancy Factor			
0 - 24 hours	N/A	100%	
24 - 96 hours		60%	
96 – 720 hours		40%	

TABLE 5-4
STEAM GENERATOR TUBE RUPTURE (SGTR)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Core Power Level	2300 MWt	2652 MWt	AST core power models planned EPU.
Initial RCS Equilibrium Activity	1.0 μCi/gm DEI-131 accident initiated iodine spike case	0.25 μCi/gm DEI-131 and 447.7 μCi/gm DEXe-133	AST values are consistent with proposed changes to more restrictive TS limits.
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DEI-131	0.1 μCi/gm DEI-131	
Maximum Pre-Accident Spike Iodine Concentration	60 μCi/gm DEI-131	60 μCi/gm DEI-131	
Iodine Spike Appearance Rate	500 times	335 times	AST – The iodine appearance rate is consistent with the guidance in RG 1.183, Appendix F.
Duration of Accident Initiated Spike	8 hours	8 hours	AST – The spike duration is consistent with the guidance in RG 1.183, Appendix F.
Integrated Break Flow and Steam Release		Attachment 3 Table 2.4-2	AST - Steam releases conservatively bound current and planned EPU conditions
Ruptured SG 0 - 30 minutes	Break Flow: 118,300 lbm Steam Release: 63,000		
30 minutes - 8.5 hours	2160 lbm		
Intact SGs 0 – 2 hours	346,00 lbm		
2 hours - 24 hours	1,936,000 lbm		

TABLE 5-4 (continued)
STEAM GENERATOR TUBE RUPTURE (SGTR)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Break Flow Flashing Fraction			AST- Break flow flashing fraction used that bounds current and planned EPU conditions
Prior to Reactor Trip	N/A	21%	
Following Reactor Trip	11%	11%	
Time of Reactor Trip	N/A	291 seconds	Current – Loss of Offsite Power assumed at event initiation. AST- Loss of Offsite Power assumed concurrent with reactor trip. This conservatively models releases from the condenser steam jet air ejector, which has higher Atmospheric Dispersion Factors than the ADVs or MSSVs.
Time to Isolate Ruptured SG	30 minutes	30 minutes	
Steam Generator Tube Leakage Rate	500 gpd/SG	0.2 gpm/SG	AST – Accident leakage consistent with analysis assumptions and proposed change to SG performance criterion TS.
Time to Establish Shutdown Cooling and Terminate Intact Steam Release	24 hours	63 hours	Cooldown times conservatively bound the current and planned EPU conditions.
Time for RCS to reach 212oF and Terminate SG Tube Leakage	24 hours	125.4 hours	
Minimum RCS Mass	366,086 lbm	366,086 lbm	Current - Minimum RCS mass bounds the current and EPU conditions due to higher steam generator tube plugging assumption (20% vs. 10%).
SG Secondary Side Mass	81,860 lbm per SG	67,707 lbm per SG	AST - The minimum mass for the intact SGs is assumed to maximize the release due to RCS iodine spiking and the value bounds current and planned EPU conditions..

TABLE 5-4 (continued)
STEAM GENERATOR TUBE RUPTURE (SGTR)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Time to Re-Cover Intact SG Tubes	N/A	30 minutes	Current – Dose consequences from tube uncovery are not part of the current licensing basis.
Tube Uncovery Flashing Fraction	N/A	11%	
Steam Generator Secondary Side Partition Coefficients			
SG (Flashed tube flow)	1.0	None	Current – values are multipliers AST – value is a divisor
SG (Non-Flashed tube flow)	0.01	100	
Condenser	N/A	100	
Atmospheric Dispersion Factors			AST - All Dispersion Factors recalculated using guidance from RG 1.194 and RG 1.145.
Offsite		Attachment 3 Table 1.8.2-1	
0 – 2 hours (EAB)	1.54E-4 m^3/sec		
0 – 2 hours (LPZ)	1.5 E-5 m^3/sec		
2 – 12 hours (LPZ)	6.5 E-6 m^3/sec		
>12 hours (LPZ)	2.4 E-7 m^3/sec		
Onsite	N/A	Attachment 3 Tables 1.8.1-2 and 1.8.1-3	Current – CR doses not calculated – bounded by LOCA analysis
Control Room Ventilation System	N/A	Attachment 3 Table 1.6.3-1	Current – CR doses bounded by LOCA analysis.
Time of Control Room Isolation	N/A	321 seconds	
Control Room Unfiltered Inleakage	N/A	115 cfm	

TABLE 5-4 (continued)
STEAM GENERATOR TUBE RUPTURE (SGTR)
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Breathing Rates				
	Offsite 0 – 8 hours	3.47 E-4 m^3/sec	3.5 E-4 m^3/sec	
	8 – 24 hours	1.75 E-4 m^3/sec	1.8 E-4 m^3/sec	
	After 24 hours	N/A	2.3 E-4 m^3/sec	
	Control Room Duration of the event	N/A	3.5 E-4 m^3/sec	
Control Room Occupancy Factor				
	0 - 24 hours	N/A	100%	
	24 - 96 hours	N/A	60%	
	96 – 720 hours	N/A	40%	

TABLE 5-5
LOCKED ROTOR
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Core Power Level	2346 MWt	2652 MWt	AST core power models planned EPU.
Core Fission Product Inventory	See Table 5-1 values	Attachment 3 Table 1.7.4-1	The AST core fission product inventory bounds current and planned EPU conditions.
Iodine Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DEI-131	0.1 μCi/gm DEI-131	
Release Fraction from Breached Fuel			
I-131	0.10	0.08	AST -RG 1.183 Section 3.2 Table 3
Kr-85		0.10	
Other Noble Gases		0.05	
Other Halogens		0.05	
Alkali Metals		0.12	
Core Average Fuel Burnup	40,000 MWD/MT	45,000 MWD/MTU	AST core average burnup conservatively increased to accommodate a planned EPU.
Fuel Enrichment	3.0 – 4.5% U-235	3.0 – 5.0 % U-235	AST maximum fuel enrichment conservatively increased to accommodate a planned EPU.
Radial Peaking Factor	1.7	1.65	AST assumed radial peaking factor combined with the planned EPU core power value bounds the current radial peaking factor combined with the current power level.
Fuel Failure	10%	15% DNB	AST - Assumed fuel failures conservatively increased to accommodate a planned EPU.

TABLE 5-5 (continued)
LOCKED ROTOR
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Minimum RCS Mass	366,086 lbm	366,086 lbm	Current - Minimum RCS mass bounds the current and EPU conditions due to higher steam generator tube plugging assumption (20% vs. 10%).
Steam Generator Tube Leakage Rate	1.0 gpm at accident conditions	0.2 gpm/SG at room temperature conditions	AST values are consistent with proposed changes to lower TS limits.
Time to Establish Shutdown Cooling and Terminate Steam Release	24 hours	63 hours	AST - Cooldown times conservatively bound the current and planned EPU conditions.
Time for RCS to reach 212°F and Terminate SG Tube Leakage	24 hours	125.4 hours	
Time to Re-Cover SG Tubes	N/A	30 minutes	Current – Dose consequences from tube uncovery are not part of the current licensing basis.
Tube Uncovery Flashing Fraction	N/A	11%	
SG Secondary Side Mass	84,128 lbm per SG	67,707 lbm per SG	AST - The minimum mass for the intact SGs is assumed to maximize the release due to RCS iodine spiking and the value bounds current and planned EPU conditions..

**TABLE 5-5 (continued)
LOCKED ROTOR
INPUTS AND ASSUMPTIONS**

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Secondary Side Mass Releases to Environment		Attachment 3 Table 2.5-2	AST - Steam releases conservatively bound current and planned EPU conditions
0 – 2 hours	521,000 lbm		
2 – 8 hours	448,400 lbm		
8 – 24 hours	1,196,000 lbm		
Secondary Side Partition Coefficients			
SG (Flashed Leakage)	N/A	None	Current – values are multipliers AST – value is a divisor
SG (Non-Flashed Leakage)	N/A	100	
Iodine Partition Factor	0.01	N/A	
Atmospheric Dispersion Factors			AST - All Dispersion Factors recalculated using guidance from RG 1.194 and RG 1.145.
Offsite		Attachment 3 Table 1.8.2-1	
0 – 2 hours (EAB)	1.54E-4 m^3/sec		
0 – 2 hours (LPZ)	1.5 E-5 m^3/sec		
2 – 12 hours (LPZ)	6.5 E-6 m^3/sec		
>12 hours (LPZ)	2.4 E-7 m^3/sec		
Onsite	N/A	Attachment 3 Tables 1.8.1-2 and 1.8.1-3	Current – CR doses bounded by LOCA analysis.
Control Room Ventilation System	N/A	Attachment 3 Table 1.6.3-1	Current – CR doses bounded by LOCA analysis.
Time of Control Room Isolation	N/A	60 seconds	
Control Room Unfiltered Inleakage	N/A	115 cfm	

TABLE 5-5 (continued)
LOCKED ROTOR
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Breathing Rates				
	Offsite 0 – 8 hours	3.47 E-4 m^3/sec	3.5 E-4 m^3/sec	
	8 – 24 hours	1.75 E-4 m^3/sec	1.8 E-4 m^3/sec	
	After 24 hours	N/A	2.3 E-4 m^3/sec	
	Control Room Duration of the event	N/A	3.5 E-4 m^3/sec	
Control Room Occupancy Factor				
	0 - 24 hours	N/A	100%	
	24 - 96 hours	N/A	60%	
	96 – 720 hours	N/A	40%	

TABLE 5-6
RCCA EJECTION
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Core Power Level	2346 MWt	2652 MWt	AST core power models planned EPU.
Core Average Fuel Burnup	40, 000 MWD/MTU	45,000 MWD/MTU	AST core average burnup conservatively increased to accommodate a planned EPU.
Fuel Enrichment	3.0 – 4.5% U-235	3.0 – 5.0 % U-235	AST maximum fuel enrichment conservatively increased to accommodate a planned EPU.
Radial Peaking Factor	1.7	1.65	AST assumed radial peaking factor combined with the planned EPU core power value bounds the current radial peaking factor combined with the current power level.
Fuel Failure	10% DNB	10% DNB / 6.37% DNB	AST – The case with lower fuel failure, secondary release and manual CR isolation was found to have limiting CR dose consequences.
Percent of Core with Centerline Melt	0.25%	0.25% / 0.16%	
Gap Release Fraction	0.10	RG 1.183 Appendix H Position 1	
Core Fission Product Inventory	See Table 5-1 values	Attachment 3 Table 1.74-1	The AST core fission product inventory bounds current and planned EPU conditions.
Initial RCS Equilibrium Activity	60 μCi/gm DEI-131	0.25 μCi/gm DEI-131 and 447.7 μCi/gm DEXe-133	AST values are consistent with proposed changes to more restrictive TS limits.
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DEI-131	0.1 μCi/gm DEI-131	

**TABLE 5-6 (continued)
RCCA EJECTION
INPUTS AND ASSUMPTIONS**

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Release from DNB Fuel			RG 1.183 Appendix H Section 1	
	I-131	6.24 E5		
	I-132	.8.96 E5		
	I-133	1.27 E6		
	I-134	1.39 E6		
	I-135	1.19 E6		
	Kr-85m	1.64 E7		
	Kr-85	9.63 E5		
	Kr-87	3.15 E7		
	Kr-88	4.44 E7		
	Xe-131m	7.29 E5		
	Xe-133m	3.99 E6		
	Xe-133	1.30 E8		
	Xe-135m	2.46 E7		
	Xe-135	3.63 E7		
	Xe-138	1.05 E8		

TABLE 5-6 (continued)
RCCA EJECTION
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Release from Fuel Centerline Melt Fuel			RG 1.183 Appendix H Section 1	
	I-131	7.72 E4		
	I-132	1.11 E5		
	I-133	1.58 E5		
	I-134	1.73 E5		
	I-135	1.48 E5		
	Kr-85m	4.10 E4		
	Kr-85	1.95 E3		
	Kr-87	7.88 E4		
	Kr-88	1.11 E5		
	Xe-131m	1.72 E3		
	Xe-133m	9.82 E3		
	Xe-133	3.16 E5		
	Xe-135m	6.15 E4		
	Xe-135	9.05 E4		
	Xe-138	2.61 E5		
Secondary Side Partition Coefficients				
	SG (Flashed Leakage)	N/A	None	Current – values are multipliers AST – value is a divisor
	SG (Non-Flashed Leakage)	N/A	100	
	Iodine Partition Factor	0.01	N/A	

TABLE 5-6 (continued)
RCCA EJECTION
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Steam Generator Tube Leakage Rate		1.0 gpm/SG	0.2 gpm/SG	AST values are consistent with proposed changes to lower TS limits.
Time to Establish Shutdown Cooling and Terminate Steam Release		24 hours	63 hours	AST - Cooldown times conservatively bound the current and planned EPU conditions.
Time to Re-Cover SG Tubes		N/A	30 minutes	Current – Dose consequences from tube uncovery are not part of the current licensing basis.
Tube Uncovery Flashing Fraction		N/A	11%	
RCS Mass		366,086 lbm	366,086 lbm	Current - Minimum RCS mass bounds the current and EPU conditions due to higher steam generator tube plugging assumption (20% vs. 10%).
SG Secondary Side Mass		84,128 lbm per SG	67,707 lbm per SG	AST - The minimum mass for the intact SGs is assumed to maximize the release due to RCS iodine spiking and the value bounds current and planned EPU conditions..
Chemical Form of Iodine Released to Containment				AST – Values are consistent with guidance given by RG 1.183, Appendix H.
	Particulate	91%	95%	
	Elemental	4%	4.85%	
	Organic	5%	0.15%	
Chemical Form of Iodine Released to Steam Generators				
	Particulate	0%	0%	
	Elemental	100%	97%	
	Organic	0%	3%	

**TABLE 5-6 (continued)
RCCA EJECTION
INPUTS AND ASSUMPTIONS**

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE		REMARKS
Atmospheric Dispersion Factors				AST - All Dispersion Factors recalculated using guidance from RG 1.194 and RG 1.145.
Offsite		Attachment 3 Table 1.8.2-1		
0 – 2 hours (EAB)	1.54 E-4 m^3/sec			
0 – 2 hours (LPZ)	1.5 E-5 m^3/sec			
2 – 12 hours (LPZ)	6.5 E-6 m^3/sec			
>12 hours (LPZ)	2.4 E-7 m^3/sec			
Onsite	N/A	Attachment 3 Tables 1.8.1-2 and 1.8.1-3		Current – CR doses not calculated – LOCA doses are bounding
Control Room Ventilation System	N/A	Attachment 3 Table 1.6.3-1		Current – CR doses bounded by LOCA analysis. AST – Two secondary side release models were analyzed; one case relies on automatic Control Room isolation and the other case relies on manual Control Room isolation (see Enclosure 3, Section 2.6.4).
Time of Control Room Isolation	N/A	Containment Release	30 seconds	
		Secondary Release	60 seconds	
		Secondary Release	30 minutes	
Control Room Unfiltered Inleakage	N/A	115 cfm		
Breathing Rates				
Offsite 0 – 8 hours	3.47 E-4 m^3/sec	3.5 E-4 m^3/sec		
8 – 24 hours	1.75 E-4 m^3/sec	1.8 E-4 m^3/sec		
After 24 hours	2.32 E-4 m^3/sec	2.3 E-4 m^3/sec		
Control Room Duration of the event	N/A	3.5 E-4 m^3/sec		Current – CR doses not calculated

TABLE 5-6 (continued)
RCCA EJECTION
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Control Room Occupancy Factor				
	0 - 24 hours	N/A	100%	
	24 - 96 hours	N/A	60%	
	96 – 720 hours	N/A	40%	
Containment Volume		1.55 E+06 ft^3	1.55 E+06 ft^3	
Containment Leakage Rate				
	0 to 24 hours	0.25% per day	0.20% (by weight)/day	AST values are consistent with proposed changes with more restrictive TS limits.
	After 24 hours	0.125% per day	0.10% (by weight)/day	
Containment Natural Deposition Coefficients				
	Instantaneous Iodine Plateout at t=0	50%	None	
	Aerosols	None	0.1 hr^{-1}	
	Elemental Iodine	5.94/hour for DF $\leq$ 100 0/hour for DF > 100	5.58 hr^{-1}	
	Organic Iodine	None	None	
Surface Area for Natural Deposition		569,219 ft^2	537,903 ft^2	AST – Total containment surface area participating in surface deposition was reduced by 25% for natural deposition of elemental iodine.

TABLE 5-6 (continued)
RCCA EJECTION
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Emergency Containment Filter Efficiency			
Start Time Delay	300 seconds	Not credited	
Number of Units	2		
Flow Rate per Unit	33, 750 cfm		
Elemental	90%		
Methyl	30%		
Particulate	95%		
Operating Time	2 hours		

TABLE 5-7
WASTE GAS DECAY TANK (WDGT) RUPTURE
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Core Power Level	2346 MWt	2652 MWt	AST core power models planned EPU.
WGDT Inventory	RCS equilibrium with 1% fuel defects equal to 55,000 curies Dose Equivalent Xe-133	RCS equilibrium with 1% fuel defects equal to 84,274.8 curies Dose Equivalent Xe-133	AST – Curie content value conservatively bounds the TS limit of 70,000 curies Dose Equivalent XE-133 (See Enclosure 3, Section 2.7.3).
Tank Volume	525 ft^3	525 ft^3	
Tank Leak Rate (Arbitrarily high)	Instantaneous rupture	1 E6 cfm	
Atmospheric Dispersion Factors			
Offsite		Attachment 3 Table 1.8.2-1	AST - All Dispersion Factors recalculated using guidance from RG 1.194 and RG 1.145.
0 – 2 hours (EAB)	1.54 E-4 m^3/sec		
0 – 2 hours (LPZ)	1.5 E-5 m^3/sec		
Onsite	N/A	Attachment 3 Tables 1.8.1-2 and 1.8.1-3	Current – CR doses not calculated - LOCA doses are bounding
Control Room Ventilation System	N/A	Attachment 3 Table 1.6.3-1	Current – CR doses bounded by LOCA analysis.
Time of Control Room Isolation	N/A	Not isolated	
Control Room Unfiltered Inleakage	N/A	115 cfm	
Control Room Makeup Flow	N/A	1,000 cfm	

TABLE 5-7 (continued)
WASTE GAS DECAY TANK (WDGT) RUPTURE
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Breathing Rates			
Offsite 0 – 8 hours	3.47 E-4 m^3/sec	3.5 E-4 m^3/sec	
8 – 24 hours	1.75 E-4 m^3/sec	1.8 E-4 m^3/sec	
After 24 hours	2.32 E-4 m^3/sec	2.3 E-4 m^3/sec	
Control Room Duration of the event	N/A	3.5 E-4 m^3/sec	
Control Room Occupancy Factor			
0 - 24 hours	N/A	100%	
24 - 96 hours	N/A	60%	
96 – 720 hours	N/A	40%	

TABLE 5-8 SPENT FUEL CASK DROP INPUTS AND ASSUMPTIONS				
INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Core Power Level Before Shutdown		2346 MWt	2652 MWt	AST core power models planned EPU.
Core Average Fuel Burnup		45,000 MWD/MTU	45,000 MWD/MTU	AST core average burnup conservatively increased to accommodate a planned EPU.
Fuel Enrichment		3.0 – 4.5% U-235	3.0 – 5.0 % U-235	AST maximum fuel enrichment conservatively increased to accommodate a planned EPU.
Core Fission Product Inventory		See Table 5-1 values	Attachment 3 Table 1.7.4-1	AST assumed radial peaking factor combined with the planned EPU core power value bounds the current radial peaking factor combined with the current power level.
Gap Release Fractions				AST – See Attachment 3, Section 2.8.2.
	I-131	0.12	0.12	
	Kr-85	0.30	0.30	
	Other Noble Gases	0.10	0.10	
Number of Fuel Assemblies Damaged		157	157	
Delay Before Spent Fuel Movement		1,525 hours	1,525 hours	
Release Duration		2 hours	2 hours	
Water Level Above Damaged Fuel Assembly		23 feet minimum	23 feet minimum	

TABLE 5-8 (continued)
SPENT FUEL CASK DROP
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION		CURRENT VALUE	AST VALUE	REMARKS
Iodine Decontamination Factors				AST – Iodine decontamination factors consistent with RG 1.183, Appendix B were adapted for this event.
	Elemental	133	285	
	Methyl	1	N/A	
	Organic	N/A	1	
Noble Gas Decontamination Factor		1	1	
Chemical Form of Iodine in Pool				AST – The guidance from RG 1.183, Appendix B was adapted for this event.
	Elemental	99.75%	99.85%	
	Methyl	0.25%	0%	
	Organic	0%	0.15%	
Control Room Ventilation System		N/A	Attachment 3 Table 1.6.3-1	Current – CR doses not calculated - bounded by LOCA analysis.
Time of Control Room Isolation		N/A	30 minutes	
Control Room Unfiltered Inleakage		N/A	115 cfm	

TABLE 5-8 (continued)
SPENT FUEL CASK DROP
INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	CURRENT VALUE	AST VALUE	REMARKS
Chemical Form of Iodine in Pool			AST – The guidance from RG 1.183, Appendix B was adapted for this event.
Atmospheric Dispersion Factors			AST - All Dispersion Factors recalculated using guidance from RG 1.194 and RG 1.145.
Offsite		Attachment 3 Table 1.8.2-1	
0 – 2 hours (EAB)	1.54E-4 m^3/sec		
0 – 2 hours (LPZ)	1.5 E-5 m^3/sec		
Control Room	N/A	Attachment 3 Tables 1.8.1-2 and 1.8.1-3	
Breathing Rates			
Offsite 0 – 8 hours	3.47 E-4 m^3/sec	3.5 E-4 m^3/sec	
8 – 24 hours	N/A	1.8 E-4 m^3/sec	
After 24 hours		2.3 E-4 m^3/sec	
Control Room Duration of the event	N/A	3.5 E-4 m^3/sec	
Control Room Occupancy Factor			
0 - 24 hours	N/A	100%	
24 - 96 hours	N/A	60%	
96 – 720 hours	N/A	40%	

LICENSE AMENDMENT REQUEST

**ALTERNATIVE SOURCE TERM
AND CONFORMING AMENDMENT
LAR 196**

ATTACHMENT 6

REGULATORY INFORMATION SUMMARY (RIS) 2006-04

**EXPERIENCE WITH IMPLEMENTATION OF
ALTERNATIVE SOURCE TERMS**

COMPLIANCE TABLE

RIS 2006-04 ISSUE	ADDRESSED BY
1. Level of Detail Contained in LARs An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations.	The LAR provides a summary of the changes, including the justification for each change. Attachment 3 provides the AST Licensing Technical Report, including specific details on the individual radiological analyses in Section 2.0, including AST analysis input values. Attachment 5 provides a comparison of current Licensing basis analysis input parameter values with those used for the AST analyses. ARCON96 meteorological data files are being submitted on CDs under separate cover letter
2. Main Steam Isolation Valve (MSIV) Leakage and Fission Product Deposition in Piping For calculation of aerosol settling velocity in the main steam line (MSL) piping of boiling water reactors, some LARs reference Accident Evaluation Report (AEB) 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term" (Ref. 2). This is acceptable. However, it is important to note that the report was written based on the parameters of a particular plant and, therefore, the removal rate constant is specific to that plant. Any licensee who chooses to reference these AEB 98-03 assumptions should provide appropriate justification that the assumptions are applicable to their particular design.	This item is applicable only to boiling water reactors. PTN Units 3 and 4 are pressurized water reactors, therefore this item is not applicable to PTN.

RIS 2006-04 ISSUE	ADDRESSED BY
3. Control Room Habitability	
When implementing an AST, some licensees have proposed that certain engineered safety features (ESF) ventilation systems not be credited as a mitigation feature in response to an accident. In some cases, the licensee's revised design basis analysis introduced the assumption that normal (non-ESF) ventilation systems are operating during all or part of an accident scenario. Such an assumption is inappropriate unless the non-ESF system meets certain qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. For example, credit for the operation of non-ESF ventilation systems should not be assumed unless they have a source of emergency power. In addition, the operation of ventilation systems establishes certain building or area pressures based upon their flow rates. These pressures affect leakage and infiltration rates which ultimately affect operator dose. Therefore, to credit the use of these systems, licensees should incorporate the systems into the ventilation filter testing program in Section 5 of the TS. In summary, use of non-ESF ventilation systems during a DBA should not be assumed unless the systems have emergency power and are part of the ventilation filter testing program in Section 5 of the TS.	The Control Room Emergency Ventilation System is credited in the control room dose analyses. The system is classified safety related. The fans, dampers, and filter are tested in accordance with Technical Specifications 3/4 7.5. The control room ventilation outside air intakes are being relocated to new locations that fall within diverse wind sectors for post-accident contaminants. The relocated intakes will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR 50 Appendix A, GDC 19. This will reduce the potential dose to the personnel in the control room following a design basis event. Attachment 3 Section 1.6.3 describes the operation of the control room ventilation system. No credit is taken for the Auxiliary Building Ventilation or Fuel Handling Building Ventilation Systems.
Generic Letter (GL) 2003-01, "Control Room Habitability" (Ref. 5) requested licensees to confirm the ability of their facility's control room to meet applicable habitability regulatory requirements. In addition, licensees were requested to confirm that control room habitability systems were designed, constructed, configured, operated and maintained in accordance with the facility's design and licensing bases.	The response to GL 2003-01 confirmed that the PTN control room envelope and ventilation systems meet all applicable habitability regulations and are designed, constructed, configured, operated, and maintained in accordance with the current licensing basis as described in the Turkey Point UFSAR.

RIS 2006-04 ISSUE	ADDRESSED BY
4. Atmospheric Dispersion	
Licensees may continue to use atmospheric relative concentration (χ/Q) values and methodologies from their existing licensing-basis analyses when appropriate. Licensees also have the option to adopt the generally less conservative (more realistic) updated NRC staff guidance on determining χ/Q values in support of design basis control room radiological habitability assessments provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Ref. 6). Regulatory positions on χ/Q values for offsite (i.e., exclusion area boundary and low population zone) accident radiological consequence assessments are provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Ref. 7). Based on submittal reviews, the NRC staff identified the following areas of improvement for licensee submittals that propose revision of the design basis atmospheric dispersion analyses for implementing AST. They should include the following information:	Meteorological data over a five-year period (2003 through 2007) was used in the development of the new onsite X/Q factors used in the AST analyses. The meteorological data was provided in annual joint frequency distribution format for 2005 through 2007 for the offsite analysis. Attachment 3 Section 1.8 describes the methodology used to develop the PTN X/Q values. The analysis of the atmospheric dispersion coefficients used in the AST analyses was performed following the guidance in RG 1.145 and RG 1.194.
• A site plan showing true north and indicating locations of all potential accident release pathways and control room intake and unfiltered inleakage pathways (whether assumed or identified during inleakage testing).	A site plan indicating true north and locations of releases and receptor points is included in Attachment 3 as Figure 1.8.1-1.
• Justification for using control room intake χ/Q values for modeling the unfiltered inleakage, if applicable.	Justification for X/Q values for unfiltered Control Room inleakage are provided in Section 4.1.3.
A copy of the meteorological data inputs and program outputs along with a discussion of assumptions and potential deviations from staff guidelines. Meteorological data input files should be checked to ensure quality (e.g., compared against historical or other data and against the raw data to ensure that the electronic file has been properly formatted, any unit conversions are correct, and invalid data are properly identified).	Meteorological data and release-receptor distances are being provided under a separate cover letter. Assumptions and atmospheric relative concentrations are included in Attachment 3 Section 1.8. The electronic data has been verified to be properly converted and formatted.

RIS 2006-04 ISSUE	ADDRESSED BY
5. Modeling of ESF Leakage ESF systems that recirculate sump water outside the primary containment may leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (e.g., refueling water storage tank). Appendix A to RG 1.183, Regulatory Position 5, states that "the radiological consequences from the postulated [ESF] leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the [loss-of-coolant accident] LOCA."	The Emergency Core Cooling System (ECCS) leakage to the auxiliary building is 4,650 cc/hr based doubling the current licensing basis value of 2,325 cc/hr. The leakage is assumed to start at 15 minutes into the event and continue throughout the 30-day period. This portion of the analysis assumes that 10% of the total iodine is released from the leaked liquid. The form of the released iodine is 97% elemental and 3% organic. The ECCS backleakage to the RWST is assumed to be 0.1 gph based upon doubling the expected total seat leakage through both sets of motor operated valves which isolate the recirculation flow from the RWST. The leakage is assumed to start at 15 minutes into the event and continue throughout the 30-day period. The fraction of the total iodine in the RWST which becomes elemental is a function of the RWST pH and the total iodine concentration. The amount of elemental iodine in the RWST fluid which then enters the RWST air space is a function of the temperature dependent iodine partition coefficient. Attachment 3 Section 2.1.3 discusses the ECCS leakage assumptions.

RIS 2006-04 ISSUE	ADDRESSED BY
6. Release Pathways Changes to the plant configuration associated with an LAR (e.g., an "open" containment during refueling) may require a re-analysis of the design basis dose calculations. A request for TS modifications allowing containment penetrations (i.e., personnel air lock, equipment hatch) to be open during refueling cannot rely on the current dose analysis if this analysis has not already considered these release pathways. RG 1.194, Regulatory Position 3.2.4.2 supports review of penetration pathways, by stating that "leakage is more likely to occur at a penetration, [and that the] analysts must consider the potential impact of leakage from building penetrations exposed to the environment." Therefore, releases from personnel air locks and equipment hatches exposed to the environment and containment purge releases prior to containment isolation need to be addressed.	The fuel handling accident analysis considers both a dropped fuel assembly inside the containment with the equipment hatch open and an assembly drop inside the fuel handling building without credit for filtration of the fuel handling building exhaust (Attachment 3 Section 2.2.1). The spent fuel cask drop analysis considers the drop of a cask into the spent fuel pool as described in UFSAR Section 14.2.1.3. RG 1.183 does not provide any specific guidance for this event; therefore, the requirements of the fuel handling accident in Appendix B of RG 1.183 are followed (Attachment 3 Section 2.8.1) The atmospheric dispersion coefficients using the unique event release-receptor pathways are calculated for all AST analyzed events consistent with the guidance in RG 1.145 and RG 1.194.
7. Primary to Secondary Leakage Some analysis parameters can be affected by density changes that occur in the process steam. The NRC staff continues to find errors in LAR submittals concerning the modeling of primary to secondary leakage during a postulated accident. This issue is discussed in Information Notice (IN) 88-31, "Steam Generator Tube Rupture Analysis Deficiency," (Ref. 11) and Item 3.f in RIS 2001-19. An acceptable methodology for modeling this leakage is provided in Appendix F to RG 1.183, Regulatory Position 5.2.	The primary to secondary leakage rate assumed 0.60 gpm total through all SGs and 0.20 gpm for any one SG at room temperature conditions for all analyzed events with secondary release pathways. The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the steam generator leak rate TS. The value used in the analyses is 62.4 lbm/ft^3 (Attachment 3 Sections 2.3.2 for MSLB, 2.4.2 for SGTR, and 2.5.2 for Locked Rotor.

RIS 2006-04 ISSUE	ADDRESSED BY
8. Elemental Iodine Decontamination Factor (DF) Appendix B to RG 1.183 provides assumptions for evaluating the radiological consequences of a fuel handling accident. If the water depth above the damaged fuel is 23 feet or greater, Regulatory Position 2 states that "the decontamination factors for the elemental and organic [iodine] species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500.	The iodine decontamination factors for the PTN fuel handling accident and spent fuel cask drop event are Elemental – 285 and Organic - 1 (Attachment 3 Table 2.2-1).
9. Isotopes Used in Dose Assessments For some accidents (e.g., main steam line break and rod drop), licensees have excluded noble gas and cesium isotopes from the dose assessment. The inclusion of these isotopes should be addressed in the dose assessments for AST implementation.	Noble gas and cesium isotopes were considered in the dose assessments for all events except the WGDT accident which included only noble gases consistent with the guidance of BTP 11-5 of the SRP. See Attachment 3, Section 2.0.
10. Definition of Dose Equivalent 131 In the conversion to an AST, licensees have proposed a modification to the TS definition of dose equivalent I-131. Some have modified the definition to base it upon the thyroid dose conversion factors of International Commission on Radiation Protection (ICRP) Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation" (Ref. 12) or ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 13). Others have proposed a definition which is a combination of different iodine dose conversion factors, (e.g., RG 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR [Part] 50, Appendix I" (Ref. 14), ICRP Publication 2, Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 15). Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steam line break and steam generator tube rupture accident analyses.	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11 (FGR 11), 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestions." The RCS dose equivalent iodine source term has been calculated for the AST analyses consistent with the definition.

RIS 2006-04 ISSUE	ADDRESSED BY
11. Acceptance Criteria for Off-Gas or Waste Gas System Release As part of full AST implementation, some licensees have included an accident involving a release from their off-gas or waste gas system. For this accident, they have proposed acceptance criteria of 500 millirem (mrem) total effective dose equivalent (TEDE). The acceptance criterion for this event is that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.	The dose criteria for the PTN waste gas decay tank rupture event are 100 mrem TEDE at both the Exclusion Area Boundary and the Low Population Zone (Attachment 3 Section 2.7.5).
12. Containment Spray Mixing Some plants with mechanical means for mixing containment air have assumed that the containment fans intake air solely from a sprayed area and discharge it solely to an unsprayed region or vice versa. Without additional analysis, test measurements or further justification, it should be assumed that the intake of air by containment ventilation systems is supplied proportionally to the sprayed and unsprayed volumes in containment.	Containment spray provides coverage to 534,442 ft^3 of the total 1,550,000 ft^3 containment volume. Therefore the PTN containment atmosphere is not considered a single, well-mixed volume. The containment is divided into three regions: upper sprayed, unsprayed above the operating deck, and unsprayed below the operating deck. The mixing rates between the regions are based on a sensitivity study evaluating various combinations of containment fans and sprays to produce conservative mixing rates. The final conservative mixing rates are 990,000 cfm between the upper sprayed and upper unsprayed containment regions and 375,000 cfm between the lower unsprayed region and the upper unsprayed region (Attachment 3 Section 2.1.4).

LICENSE AMENDMENT REQUEST

ALTERNATIVE SOURCE TERM AND CONFORMING AMENDMENT LAR 196

ATTACHMENT 7

LIST OF MODIFICATIONS

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1. The Control Room Ventilation System intakes are being relocated. The relocated intakes will be designed to seismic criteria, protected from environmental effects, and will meet the requirements of 10 CFR 50 Appendix A, GDC 19. The new intakes will be located near the ground level of the southeast and northeast corners of the auxiliary building and will fall within diverse wind sectors for post-accident contaminants. Post-modification testing in accordance with the plant design modification procedures will ensure the Technical Specification pressurization flow remains constant and demonstrate the integrity of the relocated intakes.

2. Stainless steel wire-mesh baskets (two large and eight small baskets) containing sodium tetraborate decahydrate are being installed in the containment basement.

3. The Normal Containment Cooling system aluminum cooling fins are being replaced with copper cooling fins. The copper fins will create less debris than the aluminum fins, and thus reduce the possibility of debris buildup in the containment sump.

4. The Emergency Containment Filtration System is being removed from TS.