Attachment

Numerical Applications, Inc.

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AST Licensing Technical Report

Turkey Point Units 3 and 4

Revision 2

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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 — 6/24/10

Author (Mark Pope) — Date

Jim Harrell — 6/24/10

Reviewer (Jim Harrell) — Date

for Larry Wiles — June 24, 2010

NAI Management (Tom George) — Date

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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 100 cfm. The use of 100 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 100 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 which provides adequate operating margin while ensuring margin to the radiological acceptance criteria for the limiting event (LOCA).
- 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 filter efficiency for particulates is being increased to 99% while the methyl iodine penetration criterion remains at 2.5%.

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 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 Compliance with Regulatory Guidelines

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 (RIS) 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 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. An unfiltered inleakage rate of 100 cfm is assumed in all modes of operation.

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 <u>Radiation Source Terms</u>

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 Atmospheric Dispersion (*X/Q*) Factors

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 (2005 through 2009) is used in the development of the new onsite and offsite *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 and PAVAN 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 2005 through 2009 was provided in electronic format. The data from these files was manipulated within a spreadsheet for appropriate formatting for use with ARCON96 and PAVAN.

The meteorological data was screened and validated using a number of quantitative and qualitative tests. The METD (NUREG-0917) suite of programs was one method used to identify anomalous data or data trends. The raw data was also examined graphically and otherwise to identify and flag bad or missing data. These screening activities ensure that the meteorological data used in the atmospheric dispersion factor determination were of high quality.

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 2005 to 2009 data base, the meteorological data recovery rate is 98.3%. 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 specifies a 90% data recovery threshold for measuring and capturing meteorological data. Clearly, the 98.3% 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 98.3% 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 raw meteorological data was also processed into annual joint frequency distribution format for 2005 through 2009 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, "Meteorological Monitoring Programs for Nuclear Power Plants." The data for all years was sorted into wind speed bins using the guidance provided in RIS 2006-04. The total values for each stability class are then arranged 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.

Based on issues discovered with the past temperature instrument accuracy for measuring vertical temperature difference, an additional set of meteorological data was created with a bias applied to the nominal vertical temperature differences to account for additional temperature instrument inaccuracy. Atmospheric dispersion factors (X/Qs) were re-determined with the biased vertical temperature differences. For time periods and release-receptor locations where the re-determined atmospheric dispersion factors considering the vertical temperature difference biases were more conservative, these factors were substituted for those based on the nominal vertical temperature differences.

The tower height at which the wind speeds are measured is 11.58 meters above plant grade. There were 83 calm hours in the five 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 wind speed category is classified in Reg. Guide 1.23; however, the PAVAN code requires that the maximum speed for each category be input. Therefore, a 58.16-mph value is chosen as the upper limit on the fastest wind speed category to be consistent with Turkey Point meteorological data evaluations.

An additional process was performed using meteorological data 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. Although the meteorological data used to perform this average temperature evaluation does not coincide with the same time period used to develop the atmospheric dispersion factors, the calculated average temperature swing is considered to be representative of all years.

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 25.7 feet per second. That is, 95% of all of the hourly wind speeds at the MSSV release height are less than 25.7 feet per second.

The minimum exit velocities 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 128.5 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.

<u>Release Inputs</u>

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 plus 100 cfm of unfiltered inleakage.
- 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, 100 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 99% for particulates and 95% for elemental and organic iodine.

Removal Inputs

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 plus 100 cfm of unfiltered inleakage.
- 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, 100 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 99% for particulates and 95% for elemental 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 plus 100 cfm of unfiltered inleakage.

- 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, 100 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 99% for particulates and 95% for elemental 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 plus 100 cfm of unfiltered inleakage.

- 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, 100 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 99% for particulates and 95% for elemental 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 Locked Rotor

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 plus 100 cfm of unfiltered inleakage.

- 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, 100 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 99% for particulates and 95% for elemental 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.

Based on the potential for less fuel failures to not cause a high radiation signal at the control room intake monitor, a manual isolation case was also considered for the Locked Rotor event.

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 plus 100 cfm of unfiltered inleakage.
- 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, 100 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 99% for particulates and 95% for elemental 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 to 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.l 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 plus 100 cfm of unfiltered inleakage.

- 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, 100 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 99% for particulates and 95% for elemental 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 100 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).

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

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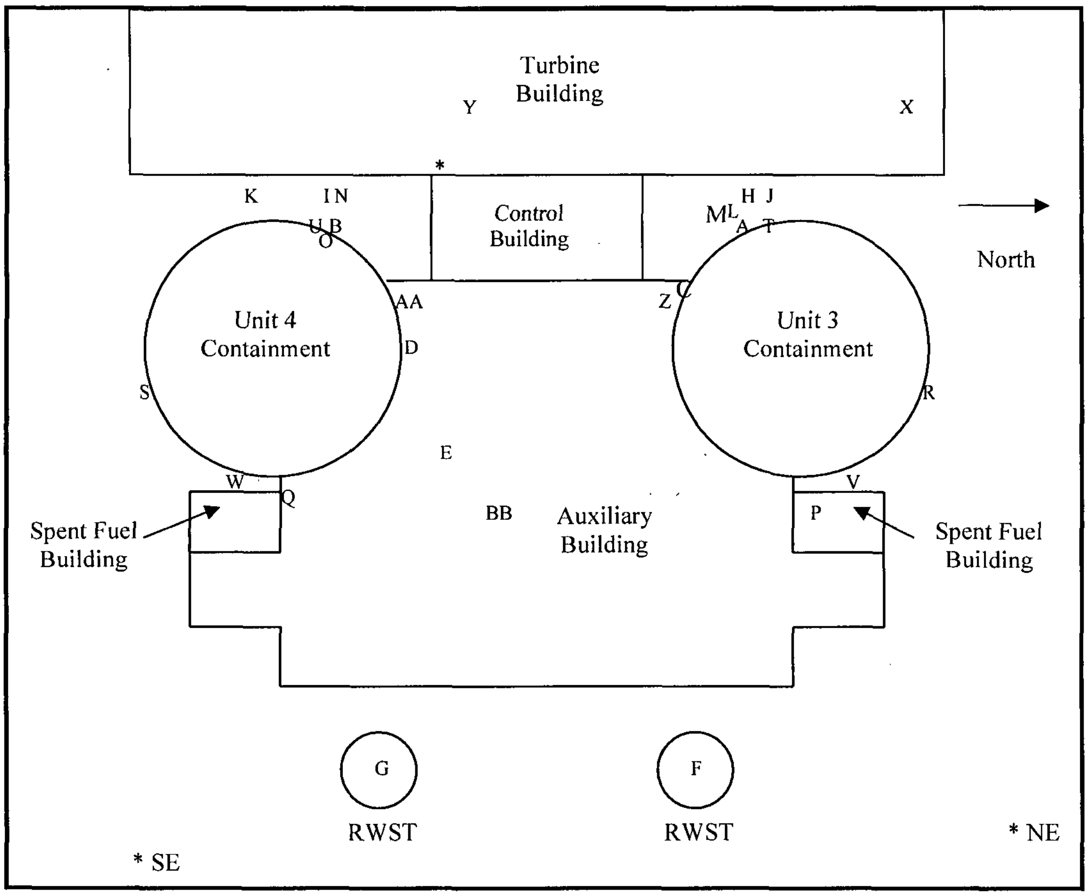
Figure 1.8.1-1

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	1000 cfm
Unfiltered Inleakage	100 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	100 cfm
Filter Efficiencies	
Elemental	95%
Organic	95%
Particulate	99%

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	1.86E-03				
B (1)	Plant stack	SE emergency intake	9.05E-04	7.62E-04	2.83E-04 (5)	2.14E-04	1.61E-04 (5)
C	Unit 4 RWST	Normal intake	9.87E-04				
D (1)	Unit 4 RWST	SE emergency intake	1.96E-03	1.55E-03	6.52E-04 (5)	4.84E-04	3.79E-04 (5)
E	Unit 4 Closest MSSV/ADV (2)	Normal intake	1.37E-02 (3)				
F (1)	Unit 4 Closest MSSV/ADV (2)	SE emergency intake	6.88E-04 (3)	4.39E-04 (5)	1.77E-04	1.28E-04	8.08E-05 (5)
G	Unit 4 Main Steam Line Closest Point	Normal intake	1.59E-02				
H (1)	Unit 4 Main Steam Line Closest Point	SE emergency intake	7.37E-04	4.57E-04 (5)	1.88E-04	1.33E-04	8.67E-05 (5)
I	Unit 4 Personnel Hatch	Normal intake	1.04E-02				
J (1)	Unit 4 Emergency Escape Lock	SE emergency intake	1.46E-03	1.06E-03 (5)	3.97E-04	3.14E-04 (5)	2.35E-04 (5)
K	Unit 4 Spent Fuel Building (NW corner)	Normal intake	2.36E-03				
L (1)	Unit 4 Spent Fuel Building (SE corner)	SE emergency intake	3.39E-03	2.77E-03	1.07E-03 (5)	8.40E-04	6.49E-04 (5)
M	Unit 4 SJAE	Normal intake	6.61E-02 (4)				
N	Unit 4 Westernmost Electrical Penetration	Normal intake	1.15E-02				
O	Auxiliary Building Vent V-10	Normal intake	2.84E-03 (5)	2.58E-03 (5)	1.28E-03 (5)	1.19E-03 (5)	8.45E-04 (5)

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 5.81E-02 sec/m^3, which is 12.1% less than the 6.61E-02 sec/m^3 value used in the analysis.

(5) The atmospheric dispersion factor calculated using the meteorological data that was adjusted to account for temperature measurement uncertainty as described in Section 1.8.3 was found to be more limiting for this case and has been applied in the dose calculation.

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	1.37E-04*	2.73E-05
0-8 hours	7.89E-05	1.23E-05
8-24 hours	6.00E-05	8.24E-06
1-4 days	3.30E-05	3.46E-06
4-30 days	1.40E-05	9.95E-07

* 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	7.103	7.108
Long Term pH	7.366	7.522

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 100 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	4.69	1.16	4.47
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 100 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	0.73	0.15	1.33
FHA in Fuel Handling Building	0.73	0.15	3.92
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 100 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	0.023	0.018	1.59
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
MSLB concurrent iodine spike	0.037	0.032	1.60
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 100 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	0.67	0.14	3.10
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	0.24	0.052	1.28
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 Time of CR Isolation (Automatic) Time of CR Isolation (Manual) Unfiltered Inleakage	Table 1.6.3-1 60 seconds - High Radiation on CR Intake Monitors 30 Minutes - Manual Isolation 100 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 – Automatic CR Isolation	0.47	0.47	1.24
Locked Rotor – Manual CR Isolation	0.076	0.076	1.25
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 - Manual Isolation 100 cfm

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	0.70	0.29	2.29
RCCA Ejection – Secondary Release (Automatic CR Isolation)	0.49	0.43	1.13
RCCA Ejection – Secondary Release (Manual CR Isolation)	0.29	0.26	3.41
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 100 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	0.066	0.013	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 100 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	0.32	0.064	2.10
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	4.69	1.16	4.47
MSLB Pre-accident Iodine Spike	0.023	0.018	1.59
SGTR Pre-accident Iodine Spike	0.67	0.14	3.10
Acceptance Criteria	**≤ 25**(3)	**≤ 25**(3)	**≤ 5**(4)
MSLB Concurrent Iodine Spike	0.037	0.032	1.60
SGTR Concurrent Iodine Spike	0.24	0.052	1.28
Locked Rotor (Automatic CR Isolation)	0.47	0.47	1.24
Locked Rotor (Manual CR Isolation)	0.076	0.076	1.25
Acceptance Criteria	**≤ 2.5** (3)	**≤ 2.5** (3)	**≤ 5** (4)
FHA – Containment Release	0.73	0.15	1.33
FHA – Fuel Building Release	0.73	0.15	3.92
Spent Fuel Cask Drop	0.32	0.064	2.10
RCCA Ejection – Containment	0.70	0.29	2.29
RCCA Ejection – Secondary (Automatic CR Isolation)	0.49	0.43	1.13
RCCA Ejection – Secondary (Manual CR Isolation)	0.29	0.26	3.41
Acceptance Criteria	**≤ 6.3**(3)	**≤ 6.3**(3)	**≤ 5**(4)
WGDT	0.066(2)	0.013	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