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Title: AST Licensing Technical Report for Palisades

Description:

This report documents the results of the analyses and evaluations performed by Numerical Applications, Inc. in support of the Palisades licensing project to implement alternative radiological source terms. Design basis accidents and radiological consequences are evaluated using the AST methodology to support control room habitability. The analyses and evaluations performed by NAI are based on the guidance of Regulatory Guide 1.183.

Revision 1 of this report is issued to incorporate the latest revisions to the accident analysis calculations and to improve the clarity of the report based on feedback received from the NRC during a pre-submittal meeting.

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1.0 Radiological Consequences Utilizing the Alternative Source Term Methodology

1.1. Introduction

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

Regulatory Guide (RG) 1.183 provides guidance on application of Alternative Source Terms (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 was 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 alternative source terms (ASTs).

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. While the Palisades tracer gas test demonstrated inleakage less than that assumed in the FSAR analyses, the AST methodology, established in RG 1.183 as supplemented by Regulatory Issue Summary 2006-04, is being used to calculate the offsite and control room radiological consequences for Palisades to support the control room habitability program by establishing a conforming set of 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)
- Small Line Break Outside Containment (SLBOC)
- Control Rod Ejection (CRE)
- Fuel Handling Accident (FHA)
- Spent Fuel Cask Drop

Note that Sections 14.7.1.3 and 14.7.1.4 of the Palisades UFSAR state that the radiological consequences of the RCP Seized Rotor event are not analyzed since all applicable acceptance criteria are met. Therefore, the Locked Rotor Accident is not included in this report. Each accident listed above, along with 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 10 cfm. The use of 10 cfm as a design basis value will be established to be above the unfiltered inleakage value determined through modification, testing and analysis consistent with the resolution of issues identified in NEI 99-03 and Generic Letter 2003-01.

1.3. Proposed Changes to the Palisades Licensing Basis

Nuclear Management Company, LLC (NMC) proposes to revise the Palisades licensing basis to implement the AST, described in RG 1.183, through reanalysis of the radiological consequences of the UFSAR Chapter 14 accidents listed in Section 1.2 above. As part of the full implementation of this AST, the following changes are assumed in the analysis:

- 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, as required.
- Dose conversion factors for inhalation and submersion are from Federal Guidance Reports (FGR) Nos. 11 and 12 respectively.

Accordingly, the following changes to the Palisades Technical Specifications (TS) are proposed:

- The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference Table 2.1 of 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.

1.4. Compliance with Regulatory Guidelines

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

- Selection of the Small Line Break Outside Containment dose consequences methodology and acceptance criteria are based on Standard Review Plan Section 15.6.2 and Regulatory Guide 1.183.

- Selection of the Spent Fuel Cask Drop dose consequences methodology and acceptance criteria are based on the Fuel Handling Accident from Regulatory Guide 1.183.

- 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.

- Use of the QADMOD-GP code to develop direct shine dose to the Control Room from the SIRWT. QAD is recommended for determining shielded dose in Standard Review Plan Section 12.3.

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	5.15	Atmospheric Dispersion Factors
MicroShield	5.05	5.16	Direct Shine Dose Calculations
QADMOD-GP	November 1999	5.41	Direct Shine Dose Calculations
ORIGEN	2.1	5.17	Core Fission Product Inventory
PAVAN	2.0	5.18	Atmospheric Dispersion Factors
RADTRAD-NAI	1.1a(QA)	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. QADMOD-GP – 3-D computer code used to analyze shielding and estimate exposure from gamma radiation.

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

1.5.5. 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.

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

RADTRAD-NAI began with versions 3.01 and 3.02 of the NRC's RADTRAD computer code, originally developed by Sandia National Laboratory (SNL). The code is initially modified to compile on a UNIX system. Once compiled, an extensive design review/verification and validation process began on the code and documentation. The subject of the review also included the source code for the solver, which is made available in a separate distribution from the NRC. RADTRAD-NAI validation is 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 is 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 1.6.4), 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 RG 1.183 and requirements of 10CFR50.67. The acceptance criteria for specific postulated accidents are provided in Table 6 of RG 1.183. For analyzed events not addressed in RG 1.183, the basis used to establish the acceptance criteria for the radiological consequences is provided in the discussion of the event in Section 2.0. For Palisades, the events not specifically addressed in RG 1.183 are the Small Line Break Outside Containment and the Spent Fuel Cask Drop.

1.6.3. Control Room HVAC System Description

The Control Room HVAC System is required to assure control room habitability. The design of the control room envelope and overall description of the Control Room HVAC System are discussed in the Palisades FSAR Section 9.8.2.

The Control Room Ventilation System consists of two 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 HVAC 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 a containment high radiation signal or a safety injection signal. Recirculation mode can also be entered manually by operator action. 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.

In the normal mode, the control room envelope is slightly pressurized relative to the surroundings with outside air continuously introduced to the control room envelope. In the recirculation (emergency) mode, the control room is pressurized at a higher rate to maintain a positive pressure differential. Makeup air for pressurization is filtered before entering the control room. The recirculated air flow is filtered by the same filters as the makeup air. If offsite power is lost, the unfiltered infiltration into the control room can occur.

The net volume of the control room envelope serviced by the Control Room HVAC System is approximately 76,451 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 CR 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 RG 1.183. Figure 1.8.1-1 provides a site sketch showing the Palisades 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 CR 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 normal mode prior to control room normal air intake and exhaust isolation, a single inlet to the control room with an unfiltered flow rate of 660 cfm is modeled (384.2 cfm if offsite power is lost). When in the emergency/recirculation mode, the control room model will define two concurrent air inlet paths representing the defined CR ventilation system air intake and the unfiltered inleakage into the CR. In the emergency/recirculation mode, outside air can enter the control room through the filtration/ventilation system from the emergency ventilation intake location. Unfiltered outside air can also enter the CR directly from various sources. Modeling of the Control Room conservatively addresses these factors as they apply to the various release locations for each analyzed event. Details of the CR modeling for each event are described in subsequent event analyses sections.

All unfiltered inleakage is assumed to enter the control room envelope via the normal intakes. Potential locations for control room envelope unfiltered inleakage include normal intake and purge exhaust isolation damper leakage, air handling unit drain leakage, and switchgear and cable spreading room emergency exhaust fan duct leakage. The potential for inleakage is due to the potential for portions of the control room envelope to be at a negative differential pressure with respect to outside air in these locations and conditions.

Equipment drains from the air handling units are connected to a common floor drain header which is routed to the normal waste system and then to the turbine building sump. The floor drains for the switchgear, cable spreading, diesel generator and battery rooms and turbine building are connected to this header. Loop seals in the air handling unit drain lines prevent the differential pressure from permitting inleakage. The loop seals are checked monthly and filled as needed to ensure the seals are operable and that no inleakage can occur.

Switchgear and cable spreading room emergency exhaust fan ducting passes through the technical support center (TSC), which is part of the control room envelope, and penetrates the TSC roof. This non-safety-related system exhausts air from the cable spreading room and 1C & 1D switchgear rooms. When the emergency exhaust fan is operating, there is a potential for the exhaust duct to leak air into to the TSC. Also, with the exhaust fan operating the differential pressure across the control room/cable spreading room boundary would increase, which would increase the leakage of air out of the control room envelope and potentially increase the amount of outside air required for pressurization. Operation of the emergency exhaust fan is not permitted when the Control Room HVAC system is in emergency mode so that no unfiltered inleakage from this path can occur.

Therefore, the only credible location for control room envelope unfiltered inleakage is the isolation dampers. The normal intake isolation damper intake ducting is closer to all release sources than the purge isolation damper exhaust ducting. For all events, the limiting train (i.e., highest) normal intake atmospheric relative concentrations are used to model the control room envelope unfiltered inleakage.

It is noted that two assumptions for the value of control room envelope unfiltered inleakage have been utilized. For the more limiting events where radiological dose margins are lower, a value of 10 cfm unfiltered inleakage is assumed. The more limiting events are the Maximum Hypothetical Accident / Loss of Coolant Accident, Main Steam Line Break, and Control Rod Ejection. For less limiting events where radiological dose margins are greater, a value of 100 cfm unfiltered inleakage is assumed. The less limiting events are the Steam Generator Tube Rupture, Small Line Break Outside Containment, Fuel Handling Accident and Spent Fuel Cask Drop.

For control room envelope unfiltered inleakage surveillance testing (i.e., tracer gas testing), the lower assumed unfiltered inleakage value of 10 cfm forms the basis for the acceptance criterion. The different inleakage assumptions are utilized to potentially limit the scope of any operability recommendation analyses in the event that control room integrity is ever called into question. For example (hypothetically), if control room envelope unfiltered inleakage is ever determined to be greater than 10 cfm but less than 100 cfm, events that assume 100 cfm inleakage would remain bounding and demonstrate radiological limits without re-analyses. The operability recommendation analyses could be limited to re-analysis of the events that assume 10 cfm inleakage, reducing the scope of the operability assessment analyses and allowing for a more timely response to Operations staff.

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:

- the radioactive material on the control room filters,
- the radioactive plume in the environment, and
- the activity in the primary containment atmosphere
- the activity in the containment purge lines
- The activity in the SIRWT

The contribution to the total dose to the operators from direct radiation sources such as the control room filters, the containment atmosphere, and the released radioactive plume were calculated for the LBLOCA event. The 30-day direct shine dose to a person in the control room, considering occupancy, is provided in Table 1.6.4-1. Note that shine doses assumed for other events conservatively bound the values presented in Table 1.6.4-1 for the LBLOCA event.

Direct shine dose is determined from five different sources to the control room operator after a postulated LOCA event. These sources are the containment, the control room make-up and recirculating air filters, the external cloud that envelops the control room, the containment purge line, and the SIRWT. 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 for all sources except the SIRWT. 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 that output the nuclide activity at a given point 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.

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 RG 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 1.6.4-1.

The QADMOD-GP code is used to determine direct shine exposure to a dose point located in the control room for the SIRWT source. QAD is recommended for shielded dose in Standard Review Plan Section 12.3.

1.7. Radiation Source Terms

1.7.1. Fission Product Inventory

The source term data to be used in performing alternative source term (AST) analyses for Palisades 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 - LOCA Source Term
Table 1.7.5-1 - Fuel Handling Accident Source Term
Table 1.7.6-1 - Spent Fuel Cask Drop Source Terms

The Palisades reactor core consists of 204 fuel assemblies. The full core isotopic inventory is determined in accordance with RG 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.

Sensitivity studies were performed to assess the bounding fuel enrichment and bounding burnup values. The assembly source term is based on 102% of original design power (2703 MW_{th}). For rod average burnups in excess of 54,000 MWD/MTU, the heat generation rate is limited to 6.3 kw/ft in accordance with RG 1.183. For non-LOCA events with fuel failures, a bounding radial peaking factor of 2.04 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 core inventory release fractions for the gap release and early in-vessel damage phases for the design basis LOCAs utilized those release fractions provided in RG 1.183, Regulatory Position 3.2, Table 2, "PWR Core Inventory Fraction Released into Containment." For non-LOCA events, the fractions of the core inventory assumed to be in the gap are consistent with RG 1.183, Regulatory Position 3.2, Table 3,"Non-LOCA Fraction of Fission Product Inventory in Gap." In some cases, the gap fractions listed in Table 3 are modified as required by the event-specific source term requirements listed in the Appendices for RG 1.183.

The following assumptions are applied to the source term calculations:

1. A conservative maximum fuel assembly uranium loading (440 kilograms) is assumed to apply to all 204 fuel assemblies in the core.

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

3. 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.

Conservatisms used in the calculation of fission product inventories include the following. Use of ORIGEN 2.1 with revised data libraries for extended fuel burnup. Use of a core thermal power corresponding to original plant design power plus 2% calorimetric uncertainty. Use of bounding maximum assembly and peak rod burnups. Use of bounding core average equilibrium cycle maximum burnup. Use of a bounding range of average assembly enrichments. Use of a bounding maximum assembly uranium loading. Neglect of decay of fission products during refueling outages.

1.7.2. Primary Coolant Source Term

The primary coolant source term for Palisades is based on operation with small defects in the cladding of fuel rods generating 1 percent of the core rated power at maximum equilibrium for the fuel cycle. Corrosion products are derived based on ANSI/ANS-18.1-1999.

The iodine activities are adjusted to achieve the Technical Specification 3.4.16 limit of 1.0 μCi/gm dose equivalent I-131 using the proposed Technical Specification definition of Dose Equivalent I-131 (DE I-131) and dose conversion factors for individual isotopes from FGR 11. The non-iodine species are adjusted to achieve the Technical Specification limit of 100/E-bar for non-iodine activities.

The dose conversion factors for inhalation and submersion are from Federal Guidance Reports Nos. 11 and 12 respectively.

The final adjusted primary coolant source term is presented in Table 1.7.2-1,"Primary Coolant Source Term."

Conservatisms used in the calculation of the primary coolant source term include the following. Use of conservative fission product inventories as noted above. Use of minimal purification flows. Use of appropriately conservative filter efficiencies. Use of corrosion product inventories derived from ANSI/ANS-18.1-1999. Normalization of dose equivalent iodine-131 to the technical specification maximum of 1.0 μCi/gm using dose conversion factors from FGR-11. Normalization of non-iodine species to the technical specification maximum of 100/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 3.7.17. 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.

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

Conservatisms used in the calculation of the secondary side source term include the following. Use of conservative fission product inventories and primary coolant source term as noted above. Scaling to the technical specification maximum of 0.1 μCi/gm using dose conversion factors from FGR-11.

1.7.4. LOCA Source Term

Per Section 3.1 of Reg. Guide 1.183, the inventory of fission products in the Palisades reactor core and available for release to the containment is based on the maximum original design power operation of the core (2703 MW_{th} which includes 2% uncertainty) 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, for the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory is used.

During a LOCA, all of the fuel assemblies are assumed to fail; therefore, the source term is based on an "average" assembly with a core average burnup of 39,300 MWD/MTU and an average assembly power* of 13.25 MW_{th}. The fuel enrichment ranges from a minimum of 3.0 w/o to a maximum value of 5.0 w/o. It is conservatively assumed that a maximum assembly uranium mass of 440,000 gm applies to all of the fuel assemblies.

*Average assembly power = (2703 MW_{th})(1 / 204 assemblies) = 13.25 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 LOCA source term is presented in Table 1.7.4-1,"LOCA Source Term."

Conservatisms used in the calculation of the LOCA source term include the following. Use of conservative fission product inventories as noted above. Selection of the maximum activities from the range of enrichments considered.

1.7.5. Fuel Handling Accident Source Term

The fuel handling accident for Palisades 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 2.04 is applied in determining the inventory of the damaged rods.

The LOCA source term is based on the activity of 204 fuel assemblies and the radial peaking factor is 2.04. Thus, based on the methodology specified in Reg. Guide 1.183, the fuel handling accident source term is derived by applying a factor of 2.04/204 to the LOCA source term and decaying for 48 hours. To ensure that the "bounding" assembly is identified, the activity of a peak burnup assembly (58,900 MWD/MTU), at 3.0 w/o , 4.0 w/o, and 5.0 w/o, is determined and compared to the source term derived from the LOCA data. For each nuclide, the bounding activity for the allowable range of enrichments and discharge exposure is determined.

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

Conservatisms used in the calculation of the fuel handling accident source term include the following. Use of conservative fission product inventories as noted above. Selection of the maximum activities from the range of enrichments considered. Selection of maximum activity from the maximum core burnup and maximum assembly burnup cases. Use of the technical specification radial peaking factor limit.

1.7.6. Spent Fuel Cask Drop Source Terms

Sections 14.11.3.1.1 of the FSAR describes three cask drop cases:

Case 1
"A cask drop onto 30 day decayed fuel with the Fuel Handling Building (FHB) Charcoal Filter operating with a conservative amount of unfiltered leakage. All "Isolable Unfiltered Leak Paths" are assumed to be isolated prior to event initiation. For scenario 1, charcoal filter bypass of 10% is assumed to exist for the entire duration of the release."

Case 2
"A cask drop onto 30 day decayed fuel with the Fuel Handling Building (FHB) Charcoal Filter operating with a conservative amount of unfiltered leakage. This scenario will determine the

maximum amount of Non-Isolatable Unfiltered Leakage allowable in order to just meet the offsite dose limits. This scenario also assumes isolation of isolatable leak paths prior to event isolation. For scenario 2, charcoal filter bypass of 17.5% is assumed to exist for the entire duration of the release."

Case 3
"A cask drop onto 90 day decayed fuel without FHB Charcoal Filter operating. For scenario 3, charcoal filter bypass of 100% is assumed for the entire duration of the release since the charcoal filters are not operating."

The cask drop source term is determined in a manner similar to that for the FHA. ORIGEN is used to determine the activities at 30 and 90 days after discharge. For each nuclide, the bounding activity for the allowable range of enrichments and discharge exposure is determined.

The Spent Fuel Cask Drop source terms are presented in Table 1.7.6-1. Note that these source terms are for a single assembly.

Conservatisms used in the calculation of the spent fuel cask drop source terms include the following. Use of conservative fission product inventories as noted above. Selection of the maximum activities from the range of enrichments considered. Selection of maximum activity from the maximum core burnup and maximum assembly burnup cases. Use of the technical specification radial peaking factor limit.

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). All of the default values in the ARCON96 code were unchanged from the code default values with the exception of the use of 0.2 for the Surface Roughness Length per Table A-2 of RG 1.194, and the use of 4.3 for the Averaging Sector Width Constant. The minimum wind speed was left at 0.5 m/s per the guidance instruction.

A number of various release-receptor combinations were considered for the control room X/Qs. These different cases were considered to determine the limiting release-receptor combinations for the various events. A ground level release was chosen for each scenario since none of the release points are 2.5 times taller than the closest solid structure as called out in Section 3.2.2 of Reference 5.21 for stack releases. The top of the containment structures is at an elevation of 782 ft. The highest release point is from the top of the plant stack, which is not 2.5 times higher than the nearby containment structure. The vertical velocity, stack flow, and stack radius terms were all set equal to zero since each case is a ground level release. The vent release option was not selected for any of the scenarios.

The actual release height was used in the cases. No credit was taken for effective release height due to plume rise; therefore, for the releases from the stacks, the release elevations were set equal to the stack top elevation. The elevation difference term was set equal to zero for each case since all elevation points are taken with respect to the same datum.

The only cases in this analysis that take credit for the building wake effect are the scenarios where the release is from the containment building and SIRWT. Some of the other scenarios have buildings between the release and receptor points, but for these cases the building wake was not credited for the sake of conservatism. Not crediting wakes was accomplished by setting the building area term equal to 0.01 m^2 as stated in Table A-2 of RG 1.194. The building area used is a conservatively determined containment cross sectional area, while the height is taken as the distance between the top of the cylinder portion of the containment structure and the highest auxiliary building roof elevation. This building area is equal to 1,405 m^2.

Figure 1.8.1-1 provides a sketch of the general layout of Palisades that has been annotated to highlight the release and receptor point locations described above. All releases are taken as ground releases per guidance provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessment 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. Direction values are corrected for "plant North" offset from "true North" by 22.745°.

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 described above.

Conservatisms used in the calculation of the onsite atmospheric relative concentrations include the following. Use of ARCON96 to calculate dispersion factors. Use of only ground level releases – no elevated or vent releases. Diffuse area releases not assumed for any release pathway. No credit for plume rise taken. Only containment building and SIRWT releases credit building wake. Use of a conservative building area when building wake is credited.

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*) for Analysis Events." 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.

The EAB distance used in each of the 16 downwind directions from the site was set at 677 m. These distance and direction combinations were chosen to be conservative, not taking credit for the larger distances to the EAB in the various primary directions. The LPZ boundary distance was set to 4820 m.

All of the releases are considered ground level releases because the highest possible release height is less than 2.5 times higher than the adjacent containment building; as described above. 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 offsite building wake term is 2,011 m^2, which 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. Release Point elevations are provided in Table 1.8.1-1, "Release-Receptor Combination Parameter for Analysis Events."

Conservatisms used in the calculation of the offsite atmospheric relative concentrations include the following. Use of PAVAN to calculate dispersion factors. Selection of maximum value for all downwind sectors for each time period and the 5% overall site dispersion factor values. Use of the minimum distance to site boundary in each downwind sector. Use of only ground level releases – no elevated releases. Use of a conservative building area for building wake credit. No exclusion of downwind sectors for sectors than extend over Lake Michigan.

1.8.3. Meteorological Data

Meteorological data over a five-year period (1999 through 2003) is used in the development of the new *X/Q* factors used in the analysis, which meets the guidance set forth in RG 1.194. The Palisades, Meteorological Monitoring Program, complies with RG 1.23; "Onsite Meteorological Programs," 1972. The Meteorological Monitoring Program is described in Section 2.5 of the Palisades UFSAR.

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. Since all of the Palisades cases use the same meteorological data file, all of the cases in this analysis have the same data recovery rate. Each of the output listings in ARCON96 files present the number of hours of data processed as 43,824 and the number of missing data hours as 179. This yields a meteorological data recovery rate of 99.6%. 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 χ/Q values. However, Regulatory Position C.5 of RG 1.23 requires a 90% data recovery threshold for measuring and capturing meteorological data. Clearly, the 99.6% valid meteorological data rate for the cases in this analysis exceeds the 90% data recovery limit set forth by RG 1.23. With a data recovery rate of 99.6% 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 Palisades site.

The meteorological data were also provided in annual joint frequency distribution format for 1999 through 2003. 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 RG 1.23. These data were provided for the five years in terms of the percentage of hours of that time period that fell into each classification category. The data for each category (i.e. wind speed, wind direction, and stability class unique combination) were converted from percent to number of hours. These hours are then input into the PAVAN code. Other information regarding the joint frequency distribution format for the PAVAN meteorological data may be found in RG 1.23.

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 primary 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 to 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 14.22.3 of the Palisades FSAR.

2.1.2. Compliance with RG 1.183 Regulatory Positions

The LOCA dose consequence analysis is consistent with the guidance provided in RG 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 RG 1.183, Regulatory Position 3.1, at 102% of core thermal power and 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 RG 1.183.

2. Regulatory Position 2 - The sump pH is controlled at a value greater than 7.0 based on the addition of tri-sodium phosphate (TSP) baskets or an alternate buffer. 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. The natural deposition removal coefficient for elemental iodine was determined to be 2.3 hr^{-1}.

   A natural deposition removal coefficient of 0.1 hr^{-1} is assumed (based on the Industry Degraded Core Rulemaking Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983) for all aerosols. Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3 ("Fission Product Cleanup System," Revision 2, December 1988) documents results from the Containment Systems Experiments. The Containment Systems Experiments examined containment atmosphere cleanup through natural transport processes. A large fraction of aerosols were deposited on the floor rather than on the walls indicating that sedimentation was a dominant removal process during the tests. IDCOR Program Technical Report 11.3 documents the following in the first sentence of the first full paragraph of page 3: "Settling of aerosols due to gravity is the dominant natural mechanism for fission product retention." The Containment Systems Experiments determined that there was significant sedimentation removal even with a relatively low aerosol concentration. The following paragraph is quoted for IDCOR Program Technical Report 11.3, Section 4.2.2.2.4 Sedimentation, pages 25 and 26:

   "Most of the available experimental data are from the CSE experiments (see Section B.1.1.4). The highest concentrations of particulates used in the CSE tests were in the range of 10^{-3} to 10^{-1} grams/m^3 whereas the initial concentration of aerosols in the RPV can be expected to be of the order of many hundreds of grams/m^3 and were this total quantity of aerosol actually to reach containment, the concentration would still be tens of grams/m^3. Consequently, particle sizes and removal rates to be expected inside the primary system would be higher than those predicted by the CSE experiments. However, even at the low CSE concentrations, very significant removals were noticed."

   Starting at a conservatively low Cesium air concentration of 10 $\mu g/m^3$ on Figure 4-2 of IDCOR Program Technical Report 11.3, the sedimentation removal coefficient was about 0.3/hr (based on data around the 8 hour time period and a 0.92 fraction of Cesium that settles to the floor based on Table 4-7. Note the initial [time 0] concentration in the experiment was only 10^{-3} grams/m^3 which was well below the expected "tens of grams/m^3" in the containment from the previous quotation). The Palisades AST analysis conservatively assumes an aerosol natural deposition rate of only 0.1/hr consistent with several prior approved AST submittals. As an example, the NRC found the 0.1 per hour aerosol removal rate to be reasonable for Kewaunee based on a study published in NUREG/CR-6189, 'A Simplified Model of Aerosol Removal by Natural Processes in Reactor containment,' and is therefore, acceptable."

Table 34 of NUREG/CR-6189 presents decontamination coefficients for design basis accident aerosol deposition. These decontamination coefficients are presented as a function of thermal power, time range and release phase. Table 36 of NUREG/CR-6189 presents correlations to model these decontamination coefficients as a function of thermal power, time range and release phase. NUREG/CR-6604 (RADTRAD: A Simplified Model for **RAD**ionuclide **T**ransport and **R**emoval **A**nd **D**ose Estimation) Table 2.2.2.1-1 presents correlations for determining the same natural deposition aerosol decontamination coefficients as a function of power, time and release phase (same as Table 36 of NUREG/CR-6189, but sums the gap and early in-vessel release phases). Thermal power is the only parameter that is varied in this table. The Palisades analyzed thermal power (2703 MWt) is greater than the Kewaunee Nuclear Power Plant analyzed thermal power (1851.3 MWt) from the above-referenced SER. The values of the decontamination coefficients in Table 2.2.2.1-1 of NUREG-6604 increase with thermal power for these two values, so use of the aerosol natural deposition rate of 0.1/hr is more conservative for Palisades than for Kewaunee.

No removal of organic iodine by natural deposition is assumed.

5. Regulatory Position 3.3– Per the current licensing basis, there is at least 90% spray coverage of the containment (Reference 5.30); therefore, the containment is treated as a single well mixed volume.

   The method used in the Palisades 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.

   As discussed in the SRP Section 6.5.2, 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 0.016667 hours (1 minute)
   - No surface deposition
   - No decay
   - No containment leakage

   The second RADTRAD-NAI case showed that a DF of 200 for elemental iodine was reached at a time greater than 2.515 hours.

A third RADTRAD-NAI case determined the time required to reach a DF of 50 for aerosol based on the peak aerosol mass from the first RADTRAD-NAI case. The third RADTRAD-NAI case included:

- Containment sprays actuated at 0.016667 hours (1 minute)
- Aerosol surface deposition credited
- No decay
- No containment leakage

The third RADTRAD-NAI case showed that a DF of 50 was reached at a time greater than 3.385 hours.

6. Regulatory Position 3.4 – Palisades does not have post-accident in-containment air filtration systems. Palisades does have containment air coolers, which are not credited for filtration or mixing.

7. Regulatory Position 3.5 – This position relates to suppression pool scrubbing in BWRs, which is not applicable to Palisades.

8. Regulatory Position 3.6 – This position relates to activity retention in ice condensers, which is not applicable to Palisades.

9. Regulatory Position 3.7 - A containment leak rate of 0.10% per day of the containment air is assumed for the first 24 hours. After 24 hours, the containment leak rate is reduced to 0.05% per day of the containment air.

10. Regulatory Position 3.8 - The purge system is not considered to be in operation at the beginning of the event. In addition, containment purge is not used after the beginning of the event for hydrogen control.

11. Regulatory Positions 4.1 through 4.6 apply to facilities with dual containment systems. As such, these positions are not applicable to Palisades.

12. Regulatory Position 5.1 - Engineered Safety Feature (ESF) systems that recirculate water outside the primary containment (ECCS systems) 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 ECCS system to the ESF rooms is taken as two times the Tech. Spec. allowable value of 0.2 gpm. The 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 SIRWT is also considered separately as two times 2.2 gpm until 2 hours when operator action is assumed to cross-tie the LPSI suction headers and eliminate backleakage through the SIRWT discharge lines. After 2 hours, the SIWRT backleakage is reduced to two times 0.025 and continues for the remainder of the 30-day duration. SIRWT backleakage is verified through the In-Service Testing program.

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 3% was determined based on the temperature of the containment sump liquid at the time recirculation begins. For ECCS leakage back to the SIRWT, the analysis demonstrates that the temperature of the leaked fluid will cool below 212°F prior to release to the SIRWT tank.

16. Regulatory Position 5.5 - The iodine available for release at the time recirculation begins is based on the expected sump pH history and temperature (see the Release Inputs in the Methodology section below). For the ECCS leakage to the auxiliary building, 10% of the total iodine in the leaked ECCS fluid is assumed to be available for release and is assumed to become airborne and leak directly to the environment from the initiation of recirculation through 30 days. For the ECCS leakage back to the SIRWT, the sump and SIRWT pH history and temperature are used to evaluate the amount of iodine that enters the SIRWT 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. For ECCS leakage into the SIRWT, the temperature and pH history of the sump and SIRWT are considered in determining the radioiodine available for release and the chemical form. Credit is taken for hold-up and dilution of activity in the SIRWT as allowed by Regulatory Position 5.6. Per the current design basis, a 50% reduction of the ECCS activity is taken for the release to the auxiliary building. No credit for holdup or dilution of ECCS leakage into the auxiliary building is taken.

18. Regulatory Position 6 – This position relates to MSSV leakage in BWRs, which is not applicable to Palisades.

19. Regulatory Position 7 - Containment purge is not considered as a means of combustible gas or pressure control in this analysis. In addition, routine containment purge is not active for this event.

2.1.3. Methodology

For this event, the Control Room ventilation system cycles through two 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 PCS piping, including the double-ended rupture of the largest piping in the PCS, or of any line connected to that system up to the first closed valve. Should a major break occur, depressurization of the PCS 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 RG 1.183, Regulatory Position 3.1, at 102% of core thermal power 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 39,300 MWD/MTU.

The leakage rate for the containment is 0.10% of the containment air weight per day. Per RG 1.183, Regulatory Position 3.7, the primary containment leakage rate is reduced by 50% at 24 hours into the LOCA to 0.05% /day based on the post-LOCA primary containment pressure history.

The ECCS leakage to the auxiliary building is 0.4 gpm based upon two times the current Tech. Spec. allowable value of 0.2 gpm. The temperature of the leakage is based on the sump temperature at and after the time recirculation begins. The leakage is assumed to start at 19 minutes (minimum time to recirculation) into the event and continue throughout the 30-day period. The maximum ECCS leakage flashing fraction is less than 10% and the minimum sump pH following the start of recirculation is 7.0; therefore, 10% of the total iodine in the leaked ECCS fluid is assumed to be released. The form of the released iodine is 97% elemental and 3% organic. Per the current design basis, a 50% reduction of the leaked ECCS activity is credited. Dilution and holdup of the ECCS leakage are not credited.

The ECCS backleakage to the SIRWT is initially assumed to be 4.4 gpm based upon doubling the previously mentioned value of 2.2 gpm. This leakage is assumed to start at 19 minutes into the event when recirculation starts and continue until 2 hours when operator action is assumed to cross-tie the LPSI suction headers and eliminate backleakage through the SIRWT discharge lines. After 2 hours, the backleakage is reduced to 2 x 0.025 gpm from the recirculation line only, 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 SIRWT for an extended period of time after recirculation begins. This time period is conservatively not credited for determining when the leakage reaches the SIRWT (i.e., the leakage is assumed to reach the SIRWT instantaneously allowing no time for radioactive decay); however, this time period is credited for determining the temperature of the leakage reaching the SIRWT. Based on sump pH history and pH control (pH is greater than 7 at the time recirculation begins), the iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form during the time of interest for this analysis. Precedent for this assumption was previously established in the revised Shearon Harris Alternative Source Term submittal dated August 17, 2001 (specifically page 2.22-11) and the associated Shearon Harris Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 107 to NPF-63 issued October 12, 2001 (specifically page 31 of the Safety Evaluation Report).

When introduced into the acidic solution of the SIRWT inventory, there is a potential for the particulate iodine to convert into the elemental form. The initial SIRWT pH is 4.5. Based upon the backleakage of sump water (pH of 7.0), the SIRWT pH slowly increases, reaching a maximum pH of 4.7 at 30 days (see Table 2.1-3). Using the time-dependent SIRWT pH, the amount of iodine converted to the elemental form in the SIRWT was determined based upon the data and equations provided in NUREG-5950. The SIRWT total iodine concentration (including stable iodine) as a function of time was determined. This iodine concentration ranged from a minimum value of 0 at the beginning of the event to a maximum value of 1.93E-05gm-atom/liter at 30 days (see Table 2.1-4). Based on the time-dependent SIRWT pH level and based on the amount of water leaked into the SIRWT from the sump, the elemental iodine fraction in the SIRWT was determined to range from 0 at the beginning of the event to a maximum of 0.149 at 30 days (see Table 2.1-6).

The elemental iodine in the liquid leaked into the SIRWT is assumed to become volatile and partition between the liquid and vapor space in the SIRWT based upon the temperature dependent partition coefficient for elemental iodine as presented in NUREG-5950. A GOTHIC model was used to determine the time-dependent SIRWT temperature (see Table 2.1-5). The time-dependent partition coefficient is provided in Table 2.1-7. The SIRWT is a vented tank; therefore, there will be no pressure transient in the air region that would affect the partition coefficient. The particulate portion of the leakage is assumed to be retained in the liquid phase of the SIRWT. Since no boiling occurs in the SIRWT, the release of the activity from the vapor space within the SIRWT is calculated based upon the displacement of air by the incoming leakage. The adjusted iodine release rate is determined as follows:

$$Adjusted\ \mathrm{Re}leaseRate = \frac{(Baseline\ Iodine\ Flow) \times (Iodine\ Volatile\ Fraction)}{(Partition\ Coefficient)}$$

where:

Baseline Iodine Flow = Volumetric flow from the sump (displacement due to backleakage)
Partition Coefficient (I_2) = Table 2.1-7
Iodine Volatile Fraction = Table 2.1-6, Elemental Iodine fraction available for release from the leaked water

The time dependent iodine release rate presented in Table 2.1-8 in then applied to the entire iodine inventory (particulate, elemental and organic) in the containment sump. The iodine released via the SIRWT air vent to the environment was effectively set to 100% elemental (the control room filters have the same efficiency for all forms of iodine).

The release point for each of the above sources is presented 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. The activity from the ECCS leakage into the Auxiliary Building is modeled as release from the Auxiliary Building via the plant stack with no filtration. The activity from the SIRWT is modeled as an unfiltered ground level release from the SIRWT.

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

- A loss of offsite power is assumed at the beginning of the event; therefore, the initial airflow to the control room is assumed to consist of 384.2 cfm of unfiltered air (air infiltration due to loss of control room ventilation).

- After the start of the event, and after the Diesel Generators restore power, the Control Room normal air intake is isolated due to a high containment pressure (or radiation) signal. After isolation of the Control Room normal air intake, the air flow distribution consists of 1413.6 cfm of filtered makeup flow through the emergency intake, 10 cfm of unfiltered inleakage, and 1413.6 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/aerosols, elemental iodine, and organic iodine.

LOCA Removal Inputs

Reduction of the airborne radioactivity in the containment by natural deposition is credited. The natural deposition removal coefficient for elemental iodine is 2.3 hr^{-1}. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols (applied after containment sprays are turned off at 10 hours). No removal of organic iodine by natural deposition is assumed.

Containment spray provides at least 90% coverage therefore, the Palisades containment building atmosphere is considered to be a single, well-mixed volume.

The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is 200 based on the maximum airborne elemental iodine concentration in the containment. The maximum airborne elemental iodine concentration is based on the release of 40 percent of the core iodine inventory. Based upon the elemental iodine removal rate of 4.8 hr^{-1}, the decontamination factor for elemental iodine reaches 200 at just over 2.515 hours.

The spray aerosol removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the aerosol removal rate of 2.3 hr^{-1}, the time for containment spray to produce an aerosol decontamination factor of 50 with respect to the containment atmosphere is just over 3.385 hours.

2.1.4. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event 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 Table 1.8.1-2 and Table 1.8.1-3.

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. These *X/Q* factors are provided in Table 1.8.2-1.

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, Version 5.05, Grove Engineering, is used 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 (for example, see Duane Arnold Energy Center submittal dated October 19, 2000 and associated NRC Safety Evaluation dated July 31, 2001). The QADMOD-GP code was used for determining the control room direct shine dose from the SIRWT. QADMOD-GP is a 3-D shielding analysis code that is suggested for use in the Standard Review Plan.

The post accident doses are the result of three distinct activity releases:

1. Containment leakage.
2. ESF system leakage into the Auxiliary Building.
3. ESF system leakage into the SIRWT.

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 ESF leakage.
2. External radioactive plume shine contribution from the containment and ESF 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.
5. A direct shine dose contribution from the activity in the containment purge lines.
6. A direct shine dose contribution from the activity collected in the SIRWT.

As shown in Table 2.1-9, 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 14.19 of the FSAR. The FSAR description of the FHA specifies a case that assumes all of the fuel rods in a single fuel assembly are damaged. In addition, a minimum water level of 22.5 feet is maintained above the damaged fuel assembly for both the containment and FHB release locations.

This analysis examined both a FHA inside the containment (with the equipment hatch open) and a FHA inside the FHB (with varying degrees of release filtration). Three FHA cases in the FHB were analyzed: these cases assumed that 10%, 34%, and 50% of the release passed through the FHB filtration system. The source term released from the overlying water pool is the same for both the FHB and the containment cases. RG 1.183 imposes the same 2-hour criteria for the release of the activity to the environment for either location. The control room was assumed to be manually isolated at 20 minutes.

The minimum water level above the damaged fuel is 22.5 feet. Since this is less than 23 feet specified in Section 2.0 of Appendix B to Reg. Guide 1.183, the decontamination factor for elemental iodine was reduced per the guidance provided in Reference 5.40. Using 22.5 feet of water along with the methodology outlined in Reference 5.40, produced an elemental iodine decontamination factor of 252. In combination with an organic iodine decontamination factor of 1.0, the reduced elemental iodine decontamination factor produces an overall iodine decontamination factor of 183.07.

2.2.2. Compliance with RG 1.183 Regulatory Positions

The FHA dose consequence analysis is consistent with the guidance provided in RG 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.

2. Regulatory Position 1.2 - The fission product release from the breached fuel is based on Regulatory Positions 3.1 and 3.2 of RG 1.183. Section 1.7 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 specified by Table 3 of RG 1.183. 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 22.5 feet is maintained above the damaged fuel assembly. Per Section 2.0 of Appendix B to Reg. Guide 1.183, Reference 5.40 was used to calculate the elemental iodine decontamination factor. A factor of 252 was calculated based on 22.5 feet of water. A decontamination factor of 1.0 was used for organic iodine. These two decontamination factors give an overall decontamination factor of 183.07 for iodine with 22.5 feet of water coverage.

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 for the FHA in containment. For the FHA in the FHB, three filtration cases were examined. These cases analyzed 10%, 34%, and 50% of the release passing through the FHB filtration system.

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 dose consequence analysis of the FHA are provided in Table 2.2-1. It is assumed that the fuel handling accident occurs at 48 hours after shutdown of the reactor per the current design basis documented in FSAR Section 14.19. 100% of the gap activity specified in Table 3 of RG 1.183 is assumed to be instantaneously released from a single fuel assembly into the pool. A minimum water level of 22.5 feet is maintained above the damaged fuel for the duration of the event. 100% 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. 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 RG 1.183. Section 1.7 discusses the development of the FHA source term, which is listed in Table 1.7.5-1.

With Control Room isolation, 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 660 cfm of unfiltered fresh air.

- 20 minutes after the start of the event, the Control Room is manually isolated. After isolation, the air flow distribution consists of 1413.6 cfm of filtered makeup flow from the outside, 100 cfm of unfiltered inleakage, and 1413.6 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, elemental iodine, and organic iodine.

2.2.4. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the location of the containment equipment hatch and plant stack and the operational mode of the control room ventilation system. These *X/Qs* are summarized in Table 1.8.1-2 and Table 1.8.1-3.

When the Control Room Ventilation System is in normal mode, the *X/Q* corresponds to the normal air intake to the control room. When the ventilation system is isolated, the limiting *X/Q* corresponds to the Control Room emergency air intake.

The EAB doses are determined using the 0-2 hour *X/Q* factors for the entire event, 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 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 a main steam line outside of containment. The radiological consequences of such an accident bound those of a MSLB inside containment. The affected steam generator (SG) rapidly depressurizes and releases the initial contents of the SG to the environment. Plant cool down is achieved via the remaining unaffected SG. This event is described in FSAR Section 14.14.3.

2.3.2. Compliance with RG 1.183 Regulatory Positions

The MSLB dose consequence analysis is consistent with the guidance provided in RG 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 – 2% of the fuel is assumed to experience DNB for the Palisades MSLB event.

2. Regulatory Position 2 – 2% of the fuel is assumed to experience DNB for the Palisades MSLB event. It was determined that the activity released from the damaged fuel will exceed that released by the two iodine spike cases; therefore, the two iodine spike cases were not analyzed.

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

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

5. Regulatory Position 5.1 - The primary-to-secondary leak rate is 0.3 gpm per SG.

6. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS (62.4 lb_m/ft^3).

7. Regulatory Position 5.3 – Based on the existing licensing basis, the primary-to-secondary leak rate is assumed to continue until the temperature of the leakage is less than 212°F at 12 hours. The release of radioactivity from the unaffected SG continues for 8 hours (time to place SDC in operation).

8. 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.

9. Regulatory Position 5.5.1 - In the faulted SG, 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 generator used for plant cooldown, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

12. Regulatory Position 5.5.2 - Any postulated leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG 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 SG 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 SG is limited by the moisture carryover from the SG. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%. No reduction in the release is assumed from the faulted SG.

15. Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for the intact SG.

2.3.3. Other Assumptions

1. RG 1.183 does not address secondary coolant activity. This analysis assumes that the equilibrium specific activity on the secondary side of the steam generators is equal to the Tech. Spec. limit of 0.1 μCi/gm Dose Equivalent I-131.

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

3. This evaluation assumes that the PCS mass remains constant throughout the MSLB event (no change in the PCS mass is assumed as a result of the MSLB or from the safety injection system).

4. The SG secondary side mass in the unaffected SG is assumed to remain constant throughout the event.

5. Releases from the faulted main steam line (and associated SG) are postulated to occur from the main steam line associated with the most limiting atmospheric dispersion factors. Releases from the unaffected SG are postulated to occur from the MSSV or ADV with the most limiting atmospheric dispersion factors.

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 affected SG rapidly depressurizes and releases the initial contents of the SG to the environment. Plant cooldown is achieved via the remaining unaffected SG.

The analysis assumes that the entire fluid inventory from the affected SG is immediately released to the environment. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 μCi/gm DE I-131 permitted by Tech. Specs. Primary coolant is also released into the affected steam generator by leakage across the SG tubes. Activity is released to the environment from the affected steam generator, as a result of the postulated primary-to-secondary leakage and the postulated activity levels of the primary and secondary coolants, until the affected steam generator is completely isolated at 12 hours (primary system temperature less than 212°F). Additional activity, due to SG tube leakage, is released via the unaffected SG via steaming from the unaffected SG MSSVs/ADVs for 8 hours, which is the time required to initiate SDC. These release assumptions are consistent with the requirements of RG 1.183.

2% of the fuel is assumed to experience DNB. The activity released by the damaged fuel exceeds that released by the two iodine spike cases; therefore, the two iodine spike cases were not analyzed. The source term for the MSLB was determined by adjusting the total core inventory (presented in Table 1.7.4-1) by the following:

- the fraction of fuel damaged (2%)

- the non-LOCA fission product gap fractions from Table 3 of RG 1.183
- the radial peaking factor of 2.04

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 660 cfm of unfiltered fresh air.

- 20 minutes after the start of the event, the Control Room is manually isolated. After isolation, the air flow distribution consists of 1413.6 cfm of filtered makeup flow from the outside, 10 cfm of unfiltered inleakage, and 1413.6 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, elemental iodine, and organic iodine.

2.3.5. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event 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 Table 1.8.1-2 and Table 1.8.1-3.

Releases from the intact SG are assumed to occur from the MSSV/ADV that produces the most limiting *X/Qs*. Releases from the faulted SG are assumed to occur from the location on a steam line that produces the most limiting *X/Qs*.

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. These *X/Q* factors are provided in Table 1.8.2-1.

The radiological consequences of the MSLB Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 2.3-3, the results of both cases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

The apparent limited margin for the MSLB control room operator dose is mitigated by noting the conservatisms used in the calculation of the main steam line break results. The assumption of 2% fuel failures is very conservative considering the thermal-hydraulic analysis of the MSLB demonstrates that no fuel failures would occur. In addition, no credit for the high velocity, buoyant release from the main steam safety valves or atmospheric dump valves is applied to the atmospheric dispersion factors for these releases. This is also a significant conservatism given the close proximity of the MSSV and ADV to the control room normal intakes (source of the control room envelope unfiltered inleakage) and resulting high probability that a significant fraction of the release bypasses the normal intakes.

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 Palisades SGTR event. This event is described in FSAR Section 14.15.

2.4.2. Compliance with RG 1.183 Regulatory Positions

The SGTR dose consequence analysis is consistent with the guidance provided in RG 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 Palisades SGTR event.

2. Regulatory Position 2 - No fuel damage is postulated to occur for the Palisades 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 Tech. Specs, which is a value of 40.0 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 Tech. Spec. value of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the PCS 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 0.3 gpm per SG.

8. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS (62.4 lb_m/ft^3).

9. Regulatory Position 5.3 - The release of radioactivity from both the affected and unaffected SGs is assumed to continue until shutdown cooling is in operation and steam release from the SGs is terminated (8 hours into the event).

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

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 - All steam generators effectively maintain tube coverage. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing for all steam generators.

    - Appendix E, Regulatory Position 5.5.2 - A portion of the primary-to-secondary ruptured tube flow through the SGTR is assumed to flash to vapor, based on the thermodynamic conditions in the reactor and secondary. The portion that flashes immediately to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment. Scrubbing of the flashed flow in the affected SG is credited. The methodologies presented in NUREG-0409 are used to determine the amount of scrubbing of the flashed flow.

    - Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage and ruptured tube flow that does not 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 SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

    - Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for this event for Palisades.

2.4.3. Other Assumptions

1. For the determination of the activity concentrations, the PCS and SG volumes are assumed to remain constant throughout both the pre-accident and the accident-induced iodine spike SGTR events.
2. For the determination of the amount of scrubbing in the affected SG, the time-dependent water level in the affected SG is used.
3. Data used to calculate the iodine equilibrium appearance rate are provided in Table 2.4-5, “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. In the unlikely event of a concurrent loss of power, the loss of circulating water through the condenser would eventually result in the loss of condenser vacuum. Valves in the condenser bypass lines would automatically close to protect the condenser thereby causing steam relief directly to the atmosphere from the ADVs or MSSVs.

A thermal-hydraulic analysis is performed to determine a conservative maximum break flow, break flashing flow, and steam release inventory through the faulted SG relief valves. In order to prevent SG overfill in the faulted SG, periodic releases via the ADVs occur from 0.5 to 8.0 hours. At 8 hours, all releases from the affected SG cease. Additional activity, based on the proposed primary-to-secondary leakage limits, is released via the unaffected SG via the ADVs until the RHR system is placed in operation (at 8 hours) to continue heat removal from the primary system.

Per the Palisades FSAR, Section 14.15, no fuel melt or clad breach is postulated for the SGTR event. Consistent with RG 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 40 μCi/gm DE I-131 permitted by Tech. Specs. The iodine activities for the pre-accident spike case are presented in Table 2.4-4. Primary coolant is released into the ruptured SG by the tube rupture and by the allowable primary-to-secondary leakage. Activity is released to the environment from the ruptured SG via direct flashing of a fraction of the released primary coolant from the tube rupture and also via steaming from the ruptured SG ADVs until the ruptured steam generator is isolated at 8 hours. The unaffected SG is used to cool down the plant during the SGTR event. Primary-to-secondary tube leakage is also postulated into the intact SG. Activity is released via steaming from the unaffected SG ADVs until the decay heat generated in the reactor core can be removed by the SDC system at 8 hours into the event. These release assumptions are consistent with the requirements of RG 1.183.

For the case of the accident-induced spike, the postulated STGR event induces an iodine spike. The PCS activity is initially assumed to be 1.0 μCi/gm DE I-131 as allowed by Tech. Specs. Iodine is released from the fuel into the PCS 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 2.4-5. The iodine activities for the accident-induced (concurrent) iodine spike case are presented in Table 2.4-6. 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 660 cfm of unfiltered fresh air.

- 20 minutes after the start of the event, the Control Room is manually isolated. After isolation, the air flow distribution consists of 1413.6 cfm of filtered makeup flow from the outside, 100 cfm of unfiltered inleakage, and 1413.6 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, elemental iodine, and organic iodine.

2.4.5. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event 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 Table 1.8.1-2 and Table 1.8.1-3.

Releases from the intact and faulted SGs are assumed to occur from the MSSV/ADV that produces the most limiting *X/Qs* when combined with the limiting applicable control room intake.

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. These *X/Q* factors are provided in Table 1.8.2-1.

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 PCS maximum pre-accident iodine spike and the accident-induced iodine spike are analyzed. As shown in Table 2.4-8, the radiological consequences of the Palisades SGTR event for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.5. Small Line Break Outside of Containment (SLBOC)

2.5.1. Background

This event examines the radiological consequences of the rupture of a letdown line in the reactor building. The activity from the letdown line rupture will be released to the environment by the auxiliary building ventilation system that exhausts via the plant stack. The current FSAR analysis of this event includes a pre-accident iodine spike case, a concurrent iodine spike case, and a case with an initial PCS activity of 1.0 μCi/gm with no iodine spike. Per Section 15.6.2 of the Standard Review Plan, for the small line break outside of containment, only the concurrent iodine spike case is required. In addition, the initial PCS activity case will be bounded by the concurrent iodine spike case; therefore, only the concurrent iodine spike case was analyzed. This event is described in Section 14.23 of the FSAR.

2.5.2. Compliance with RG 1.183 Regulatory Positions and other Assumptions

No specific guidance for this event is provided by RG 1.183. Therefore, the AST reanalysis for this event will follow Section 15.6.2 of the Standard Review Plan with appropriate modifications to maintain the intent of RG 1.183.

1. No fuel damage is postulated to occur for the Palisades SLBOC event.

2. Per SRP 15.6.2 – It is assumed that the transient associated with the SLBOC causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the Tech. Spec. value of 1.0 μCi/gm DE I-131. Iodine is assumed to be released from the fuel into the PCS at a rate of 500 times the iodine equilibrium release rate for a period of 8 hours.

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

4. Iodine releases from the break to the environment are assumed to be 97% elemental and 3% organic.

5. The temperature and pressure of the break flow (135^{o}F and 35 psia) will result in minimal flashing of the break flow. However, consistent with Section 5.5 of Appendix A to RG 1.183, a flashing fraction of 10% is conservatively used for determining the iodine release from the break.

6. 100% of the noble gases in the break flow are assumed to be released.

7. The initial PCS activity is assumed to be at the TS limit of 1.0 μCi/gm Dose Equivalent I-131and 100/E-bar gross activity.

2.5.3. Methodology

Input assumptions used in the dose consequence analysis of the SLBOC event are provided in Table 2.5-1.

This event is caused by the rupture of a letdown line in the auxiliary building. A 160 gpm break flow at 135^{o}F and 35 psia is postulated. 60 minutes are required to identify and isolate the break. No reactor trip is assumed.

Per Section 15.6.2 of the Standard Review Plan a concurrent iodine spike with a multiplier of 500 on the equilibrium iodine release rate is used to determine the iodine concentration in the released fluid. Table 2.5-2 lists the iodine release rates. Table 2.4-5 lists the inputs used to determine the equilibrium iodine release rates. The thermodynamic properties of the release flow will result in little flashing; however, per Section 5.5 of Appendix A to RG 1.183 (Assumptions on ESF System Leakage), it is conservatively assumed that 10% of the iodine in the break flow will be released to the reactor building airspace. The iodine released to the reactor building airspace is assumed to consist of 97% elemental and 3% organic iodine. Dilution, holdup, and plateout of the release in the reactor building are not credited. No filtration of the release from the reactor building is applied.

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 660 cfm of unfiltered fresh air.

- 20 minutes after the start of the event, the Control Room is manually isolated. After isolation, the air flow distribution consists of 1413.6 cfm of filtered makeup flow from the outside, 100 cfm of unfiltered inleakage, and 1413.6 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, elemental iodine, and organic iodine.

2.5.4. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event 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 Table 1.8.1-2 and Table 1.8.1-3.

The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake. When the Control Room Ventilation System is in normal mode, the *X/Q* corresponds to the normal air intake to the control room. When the ventilation system is isolated, the limiting *X/Q* corresponds to the Control Room emergency air intake. The *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 SLBOC release point (plant stack) 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 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 radiological consequences of the SLBOC 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. Control Rod Ejection (CRE)

2.6.1. Background

This event consists of the ejection of a single control rod assembly. The CRE results in a reactivity insertion that leads to a core power level increase and subsequent reactor trip. Following the reactor trip, plant cooldown is affected by steam release to the condenser. Two CRE 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 FSAR, Section 14.16.

2.6.2. Compliance with RG 1.183 Regulatory Positions

The CRE dose consequence analysis is consistent with the guidance provided in RG 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 RG 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 2.04 is applied. The release fractions provided in RG 1.183 Table 3 are adjusted to comply with the specific RG 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 SGs.

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 SGs 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 spray is not credited.

7. Regulatory Position 6.2 - The containment is assumed to leak at the proposed TS maximum allowable rate of 0.10% for the first 24 hours and 0.05% for the remainder of the event.

8. Regulatory Position 7.1 - The primary-to-secondary leak rate is 0.3 gpm per SG.

9. Regulatory Position 7.2 - The density used in converting volumetric leak rates to mass leak rates is consistent with the basis of surveillance tests used to show compliance with the SG leak rate TS (62.4 lb_m/ft^3).

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 - Compliance with Appendix E Sections 5.5 and 5.6 is discussed below:

    - Appendix E, Regulatory Position 5.5.1 – Both steam generators are used for plant cooldown. Therefore, the primary-to-secondary leakage is assumed to mix with the secondary water without flashing.

    - Appendix E, Regulatory Position 5.5.2 - None of the SG tube leakage is assumed to flash for this event.

    - Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage 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 SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.

    - Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncovery is not postulated for this event for Palisades.

2.6.3. Other Assumptions

1. RG 1.183 does not address secondary coolant activity. This analysis assumed that the equilibrium specific activity on the secondary side of the steam generators is equal to the TS limit of 0.1 μCi/gm Dose Equivalent I-131.

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

3. This evaluation assumed that the PCS mass remains constant throughout the event.

4. The SG secondary side mass in the SGs is assumed to remain constant throughout the event.

5. Steam releases from the SGs are postulated to occur from the MSSV or ADV with the most limiting atmospheric dispersion factors. For the CRE inside of containment release case, releases are assumed to leak out of the containment via the same containment release points discussed for the LOCA in Section 2.1.

2.6.4. Methodology

Input assumptions used in the dose consequence analysis of the CRE 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.

For the containment release case, 100% of the activity is released instantaneously to the containment. The releases from the containment correspond to the same leakage points discussed for the LOCA in Section 2.1. Natural deposition of the released activity inside of containment is credited. Removal of activity via containment spray is not credited.

For the secondary release case, primary coolant activity is released into the SGs by leakage across the SG tubes. The activity on the secondary side is then released via the SG MSSVs/ADVs. Additional activity based on the secondary coolant initial iodine concentration is assumed to be equal to the maximum value of 0.1 μCi/gm DE I-131 permitted by Tech Specs. Activity is released to the environment from the steam generator as a result of the postulated primary-to-secondary leakage and the postulated activity levels of the primary and secondary coolants, until the release is terminated (at 8 hours for shutdown cooling initiation). These release assumptions are consistent with the requirements of RG 1.183. The noble gas activity released by the tube leakage is assumed to be released directly to the environment without mitigation.

The CRE is evaluated with the assumption that 0.5% of the fuel experiences FCM and 14.7% of the fuel experiences DNB. The activity released from the damaged fuel corresponds to the requirements set out in Regulatory Position 1 of Appendix H to RG 1.183. A radial peaking factor of 2.04 is applied in the development of the source terms.

For this event, the Control Room ventilation system cycles through three modes of operation (the operational modes are summarized in Table 1.6.3-1):

- Loss of offsite power is assumed to occur at the beginning of the event. During this time period, the air flow distribution consists of 384.2 cfm of unfiltered air infiltration.
- Power to the control room ventilation system is restored at 90 seconds and the normal ventilation flow rate of 660 cfm of unfiltered air is restored.
- 20 minutes after the start of the event, the Control Room is assumed to be manually isolated. After isolation, the air flow distribution consists of 1413.6 cfm of filtered makeup flow from the outside, 10 cfm of unfiltered inleakage, and 1413.6 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, elemental iodine, and organic iodine.

2.6.5. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event 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 Table 1.8.1-2 and Table 1.8.1-3.

For the CRE secondary side release case, releases from the SGs are assumed to occur from the MSSV/ADV that produces the most limiting *X/Qs*. When the Control Room Ventilation System is in normal mode, the *X/Q* corresponds to the normal air intake to the control room. When the ventilation system is isolated, the limiting *X/Q* corresponds to the Control Room emergency air intake. For the CRE containment release case, the *X/Qs* for containment leakage are assumed to be identical to those for the LOCA discussed in Section 2.1.

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 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 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. Spent Fuel Cask Drop

2.7.1. Background

The purpose of this analysis is to reanalyze the radiological consequences of the cask drop accident presented in Section 14.11 of the Palisades FSAR. RG 1.183 does not provide any specific guidance for the cask drop event; therefore, the requirements of Appendix B of the RG (fuel handling accident) are followed for the cask drop reanalysis.

Three cask drop cases were analyzed:

- Case 1 with 90% of the release via the FHB filtration system, 30 days of decay, and the control room initially aligned in emergency recirculation mode.
- Case 2 with 82.5% of the release via the FHB filtration system, 30 days of decay, and the control room initially aligned in emergency recirculation mode.
- Case 3 with 0% of the release via the FHB filtration system, 90 days of decay, and no isolation of the control room.

Due to conflicting requirements outlined in Section 2.0 of Appendix B to Reg. Guide 1.183, the cask drop cases were analyzed with an elemental iodine decontamination factor of 285, which corresponds to an overall iodine decontamination factor of 200.

2.7.2. Compliance with RG 1.183 Regulatory Positions

The Cask Drop dose consequence analysis is consistent with the guidance provided in RG 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 damaged is consistent with the current design basis (73 damaged fuel 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 RG 1.183. Section 1.7.6 provides a discussion of how the Cask Drop source term is developed. A listing of the source terms is provided in Table 1.7.6-1. The gap activity available for release is specified by Table 3 of RG 1.183. This activity is assumed to be released 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 assemblies. Therefore, an overall effective decontamination factor of 200 is used for the iodine.

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 within 2 hours.

7. Regulatory Position 4.3 - No dilution is assumed.

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

2.7.3. 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.

2.7.4. Methodology

The input assumptions used in the dose consequence analysis of the Cask Drop are provided in Table 2.7-1. 100% of the gap activity specified in Table 3 of RG 1.183 is assumed to be instantaneously released from all of the fuel assemblies into the fuel pool. A minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event. 100% 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 activity released from the pool is assumed to leak to the environment over a two-hour period.

The source term meets the requirements of Regulatory Position 1 of Appendix B to RG 1.183. Section 1.7.6 discusses the development of the Cask Drop source terms for both cases, which are listed in Table 1.7.6-1.

For Cases 1 and 2, Control Room ventilation is assumed to already be in the emergency recirculation mode at the start of the event. 100 cfm of unfiltered leakage is assumed throughout the event. For Case 3, Control Room Ventilation remains unisolated. The Control Room ventilation system operational modes are summarized in Table 1.6.3-1). The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, elemental iodine, and organic iodine. For the cases listed above which credit filtration by the Fuel Handling Building charcoal filters, efficiencies that are applied to the release are 94% for the elemental and organic iodine. All of the particulates are filtered out by the water in the pools.

2.7.5. Radiological Consequences

The atmospheric dispersion factors (*X/Qs*) used for this event for the Control Room dose are based on the release path from the FHB and the pathway into the control room. These *X/Qs* are summarized in Table 1.8.1-2 and Table 1.8.1-3.

For the EAB dose calculation, the 0-2 hour *X/Q* factors are used for the entire event, and for the LPZ dose calculation, the *X/Q* factors from Table 1.8.2-1 are used.

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 2.7-2, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

2.8. Environmental Qualification (EQ)

The Palisades FSAR, Appendix 7C, discusses equipment EQ due to a radiation environment. RG 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. The Palisades EQ analyses will continue to be based on TID-14844 assumptions at this time.

3.0 Summary of Results

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

4.0 Conclusion

Full implementation of the Alternative Source Term methodology, as defined in Regulatory Guide 1.183 and Regulatory Issue Summary 2006-04, into the design basis accident analysis has been made to support control room habitability. 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), Small Line Break Outside Containment (SLBOC), Control Rod Ejection (CRE) Ejection, and Spent Fuel Cask Drop have been made using the RG 1.183 methodology. The analyses used assumptions consistent with proposed changes in the Palisades 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 10 cfm.

5.0 References

5.1 Palisades FSAR through Revision 25.

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,” 1989.

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

5.9 Palisades Technical Specifications through Amendment 205.

5.10 Shearon Harris Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 107 to NPF-63 issued October 12, 2001.

5.11 Crystal River Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 199 to DPR-72 issued September 17, 2001.

5.12 Fort Calhoun Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 201 to DPR-40 issued December 5, 2001.

5.13 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.

5.14 Duane Arnold Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 240 to DPR-49 issued July 31, 2001.

5.15 ARCON96 Computer Code ("Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev. 1, May 1997, RSICC Computer Code Collection No. CCC-664 and July 1997 errata).

5.16 MicroShield Version 5 "User's Manual "and "Verification & Validation Report, Rev. 5," Grove Engineering, both dated October 1996.

5.17 Oak Ridge National Laboratory, CCC-371, "RSICC Computer Code Collection – ORIGEN 2.1," May 1999.

5.18 "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).

5.19 Numerical Applications Inc., NAI-9912-04, Revision 4, "RADTRAD-NAI Version 1.1a(QA) Documentation," October 2004.

5.20 Numerical Applications Inc., "Dose Methodology Quality Assurance Procedures," Revision 1, June 4, 2001.

5.21 NRC Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," June 2003.

5.22 USNRC, Regulatory Issue Summary 2006-04, Experience With Implementation of Alternate Source Terms, March 2006..

5.23 USNRC, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, February 1983.

5.24 USNRC, Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

5.25 Letter from A. C. Bakken, III to USNRC, "Donald C. Cook Nuclear Plant Units 1 and 2, Partial Response to Second Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for Control Room Habitability," January 15, 2002.

5.26 D. C. Cook Issuance of Amendments (IA) and Safety Evaluation (SE) for Amendment No. 271 to DPR-58 and Amendment No. 252 to DPR-74 issued November 14, 2002.

5.27 NUREG-0800, USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” September 1981 (or updates of specific sections).

5.28 Industry Degraded Core Rulemaking Program Technical Report 11.3, “Fission Product Transport in Degraded Core Accidents,” Atomic Industrial Forum, December 1983.

5.29 Indian Point Nuclear Generating Unit No. 2 Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 211 to DPR-26 issued July 27, 2000.

5.30 Letter from A. Schwencer (NRC) to D. Bixel (CPCo), “Transmittal of Amendment No. 31 and Safety Evaluation”, November 1, 1977.

5.31 Reference deleted.

5.32 Fort Calhoun Issuance of Amendment (IA) and Safety Evaluation (SE) for Amendment No. 198 to DPR-40 issued April 4, 2001.

5.33 USNRC, Regulatory Guide 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Rev. 3, June 2001.

5.34 NRC Generic Letter 99-02, “Laboratory Testing of Nuclear-Grade Activated Charcoal,” June 3, 1999.

5.35 NRC Information Notice 91-56, “Potential Radioactive Leakage to Tank Vented to Atmosphere,” September 19, 1991.

5.36 Letter from J. Scarola to USNRC, “Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Revision to Alternate Source Term Methodology Analyses in Support of the Steam Generator Replacement and Power Uprate License Amendment Applications,: August 17, 2001.

5.37 NUREG/CR-5950, “Iodine Evolution and pH Control,” December 1992.

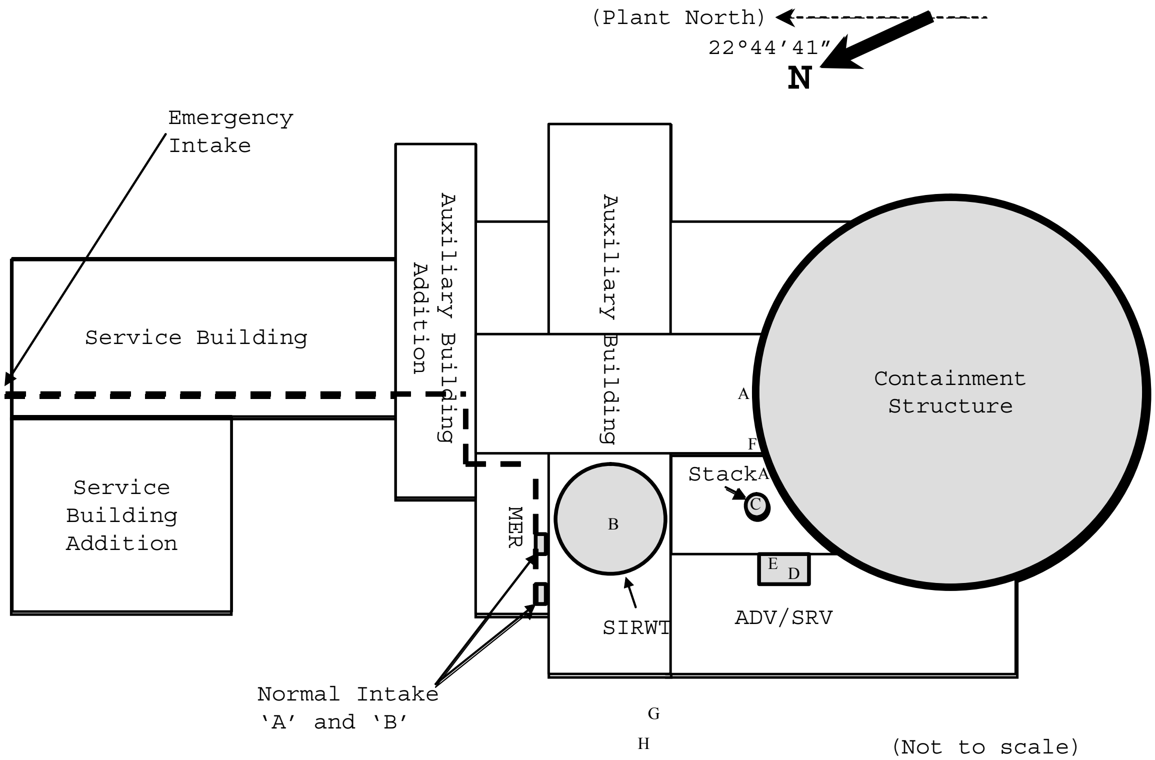
5.38 Reference deleted.

5.39 NAI Calculation Number NAI-1101-002 Rev. 0, “Qualification of ORIGEN2.1 for Florida Power & Light AST Applications,” August 8, 2002.

5.40 G. Burley, “Evaluation of Fission Product Release and Transport,” Staff Technical Paper, 1971.

5.41 ORNL RSICC Code Package CCC-565, “QADMOD-GP: Point Kernel Gamma-Ray Shielding Code with Geometric Progression Buildup Factors”, November 1990.

Figure 1.8.1-1 Onsite Release-Receptor Location Sketch



* – Control Room Intakes / Receptor Point
A – Containment Closest Point
B – SIRWT Vent
C – Plant Stack
D – Closest ADV
E – Closest SSRV
F – Equipment Door
G – Turbine Building NE Roof Exhauster
H – Turbine Building NW Roof Exhauster

Table 1.6.3-1 Control Room Ventilation System Parameters

Parameter	**Value**
Control Room Volume	35,923 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate and Inleakage	660 cfm 384.2 cfm during periods without offsite power
Emergency Operation	
Recirculation Mode:	
Filtered Make-up Flow Rate	1413.6 cfm
Filtered Recirculation Flow Rate	1413.6 cfm
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage (LOCA, MSLB, CRE)	10 cfm
Unfiltered Inleakage (SGTR, FHA, Cask Drop, Small Line Break)	100 cfm
Filter Efficiencies	
Elemental	99%
Organic	99%
Particulate	99%

Table 1.6.4-1 LOCA Direct Shine Dose

Source	Direct Shine Dose (rem)
Containment	0.028
CR Filters	0.005
Purge Line	0.000
External Cloud	0.232
SIRWT	1.269
Total	**1.534**

Table 1.7.2-1 Primary Coolant Source Term

Nuclide	μCi/gm	Nuclide	μCi/gm
KR-83M	2.349E-01	BA-140	5.211E-03
KR-85M	8.808E-01	LA-140	2.496E-03
KR-85	3.890E-01	CE-141	8.074E-04
KR-87	5.505E-01	CE-143	4.624E-04
KR-88	1.615E+00	CE-144	6.826E-04
XE-131M	3.303E-01	PR-143	7.340E-04
XE-133M	1.248E+00	MN-54	1.762E-03
XE-133	5.138E+01	FE-55	1.321E-03
XE-135	5.432E+00	CO-58	5.138E-03
BR-83	4.844E-02	ZR-97	3.597E-04
BR-84	1.982E-02	RU-105	1.028E-04
RB-86	1.101E-02	RH-105	4.037E-04
CS-134	3.597E+01	SB-127	3.156E-02
CS-136	3.156E+00	SB-129	1.762E-02
CS-137	1.908E+01	XE-135M	3.523E-01
CS-138	5.138E-01	CS-134M	3.376E-02
SR-89	3.964E-03	RB-88	1.688E+00
SR-90	3.597E-04	RB-89	4.257E-02
CR-51	3.450E-03	SB-124	8.074E-04
FE-59	3.303E-04	SB-125	6.753E-03
CO-60	5.872E-04	SB-126	4.551E-04
SR-91	1.101E-03	TE-131	1.101E-02
SR-92	4.331E-04	TE-133M	8.808E-03
Y-90	4.698E-04	BA-141	6.679E-05
Y-91M	6.533E-04	BA-139	3.156E-04
Y-91	1.321E-02	LA-141	1.762E-04
Y-92	5.285E-04	LA-142	4.991E-05
Y-93	3.230E-04	ND-147	3.156E-04
ZR-95	8.808E-04	NB-97	5.945E-05
NB-95	8.808E-04	NB-95M	6.239E-06
MO-99	3.083E+00	PM-147	8.808E-05
TC-99M	2.349E+00	PM-148	1.248E-04
RU-103	8.808E-04	PM-149	2.202E-04
RU-106	3.890E-04	PM-151	6.092E-05
RH-103M	8.808E-04	PM-148M	1.908E-05
TE-125M	1.468E-03	PR-144	6.826E-04
TE-127M	5.285E-03	Y-94	1.028E-05
TE-127	3.376E-02	BR-82	1.835E-02

TE-129M	1.615E-02	I-131	0.8305
TE-129	2.789E-02	I-132	0.1917
TE-131M	2.716E-02	I-133	0.8624
TE-132	3.523E-01	I-134	0.0751
TE-134	1.541E-02	I-135	0.3673

* The iodine activities have been adjusted to the Tech. Spec. limit of 1.0 μCi/gm DE I-131. Non-iodine activities have been adjusted to the Tech. Spec. limit of 100/E-bar.

Table 1.7.3-1 Secondary Side Source Term

Isotope	μCi/gm
I-131	0.0830
I-132	0.0192
I-133	0.0862
I-134	0.0075
I-135	0.0367

Table 1.7.4-1 LOCA Source Term

Nuclide	Curies	Nuclide	Curies
Co-58	0.0000E+00	Pu-239	0.3558E+05
Co-60	0.0000E+00	Pu-240	0.5406E+05
Kr-85	0.1052E+07	Pu-241	0.1522E+08
Kr-85m	0.1948E+08	Am-241	0.1884E+05
Kr-87	0.3756E+08	Cm-242	0.5669E+07
Kr-88	0.5286E+08	Cm-244	0.5943E+06
Rb-86	0.1959E+06	I-130	0.3743E+07
Sr-89	0.7213E+08	Kr-83m	0.9119E+07
Sr-90	0.8458E+07	Xe-138	0.1211E+09
Sr-91	0.8874E+08	Xe-131m	0.8346E+06
Sr-92	0.9557E+08	Xe-133m	0.4659E+07
Y-90	0.8737E+07	Xe-135m	0.2999E+08
Y-91	0.9264E+08	Cs-138	0.1340E+09
Y-92	0.9596E+08	Cs-134m	0.4920E+07
Y-93	0.1101E+09	Rb-88	0.5369E+08
Zr-95	0.1236E+09	Rb-89	0.6895E+08
Zr-97	0.1206E+09	Sb-124	0.1702E+06
Nb-95	0.1249E+09	Sb-125	0.1567E+07
Mo-99	0.1368E+09	Sb-126	0.1107E+06
Tc-99m	0.1198E+09	Te-131	0.6601E+08
Ru-103	0.1260E+09	Te-133	0.8639E+08
Ru-105	0.9451E+08	Te-134	0.1220E+09
Ru-106	0.5794E+08	Te-125m	0.3413E+06
Rh-105	0.8741E+08	Te-133m	0.5406E+08
Sb-127	0.9111E+07	Ba-141	0.1188E+09
Sb-129	0.2568E+08	Ba-137m	0.1043E+08
Te-127	0.9047E+07	Pd-109	0.3327E+08
Te-127m	0.1223E+07	Rh-106	0.6285E+08
Te-129	0.2528E+08	Rh-103m	0.1135E+09
Te-129m	0.3772E+07	Tc-101	0.1261E+09
Te-131m	0.1113E+08	Eu-154	0.1247E+07
Te-132	0.1048E+09	Eu-155	0.8448E+06
I-131	0.7483E+08	Eu-156	0.2023E+08
I-132	0.1068E+09	La-143	0.1108E+09
I-133	0.1462E+09	Nb-97	0.1216E+09
I-134	0.1602E+09	Nb-95m	0.8835E+06
I-135	0.1372E+09	Pm-147	0.1292E+08

Nuclide	Curies	Nuclide	Curies
Xe-133	0.1466E+09	Pm-148	0.2144E+08
Xe-135	0.4692E+08	Pm-149	0.4541E+08
Cs-134	0.2037E+08	Pm-151	0.1606E+08
Cs-136	0.5873E+07	Pm-148m	0.2999E+07
Cs-137	0.1100E+08	Pr-144	0.1025E+09
Ba-139	0.1307E+09	Pr-144m	0.1224E+07
Ba-140	0.1260E+09	Sm-153	0.4423E+08
La-140	0.1299E+09	Y-94	0.1105E+09
La-141	0.1193E+09	Y-95	0.1183E+09
La-142	0.1156E+09	Y-91m	0.5151E+08
Ce-141	0.1212E+09	Br-82	0.5282E+06
Ce-143	0.1115E+09	Br-83	0.9102E+07
Ce-144	0.1020E+09	Br-84	0.1591E+08
Pr-143	0.1111E+09	Am-242	0.9062E+07
Nd-147	0.4770E+08	Np-238	0.4306E+08
Np-239	0.1830E+10	Pu-243	0.4690E+08
Pu-238	0.3927E+06		

Table 1.7.5-1 Fuel Handling Accident Source Term

Nuclide	Bounding Activities (Curies)	Nuclide	Bounding Activities (Curies)	Nuclide	Bounding Activities (Curies)
Co-58	0.0000E+00	I-135	0.8949E+04	Sb-126	0.9900E+03
Co-60	0.0000E+00	Xe-133	0.1298E+07	Te-131	0.8307E+04
Kr-85	0.1052E+05	Xe-135	0.8201E+05	Te-133	0.2034E-10
Kr-85m	0.1174E+03	Cs-134	0.2034E+06	Te-134	0.2217E-14
Kr-87	0.1647E-05	Cs-136	0.5284E+05	Te-125m	0.3417E+04
Kr-88	0.4302E+01	Cs-137	0.1100E+06	Te-133m	0.1213E-09
Rb-86	0.1819E+04	Ba-139	0.4861E-04	Ba-141	0.0000E+00
Sr-89	0.7020E+06	Ba-140	0.1130E+07	Ba-137m	0.1041E+06
Sr-90	0.8456E+05	La-140	0.1235E+07	Pd-109	0.2825E+05
Sr-91	0.2679E+05	La-141	0.2730E+03	Rh-106	0.5771E+06
Sr-92	0.4453E+01	La-142	0.5794E-03	Rh-103m	0.1097E+07
Y-90	0.8623E+05	Ce-141	0.1168E+07	Tc-101	0.0000E+00
Y-91	0.9107E+06	Ce-143	0.4100E+06	Eu-154	0.1246E+05
Y-92	0.3229E+03	Ce-144	0.1014E+07	Eu-155	0.8442E+04
Y-93	0.4137E+05	Pr-143	0.1071E+07	Eu-156	0.1935E+06
Zr-95	0.1210E+07	Nd-147	0.4211E+06	La-143	0.0000E+00
Zr-97	0.1684E+06	Np-239	0.1023E+08	Nb-97	0.1692E+06
Nb-95	0.1248E+07	Pu-238	0.4494E+04	Nb-95m	0.8748E+04
Mo-99	0.8264E+06	Pu-239	0.3578E+03	Pm-147	0.1296E+06
Tc-99m	0.7956E+06	Pu-240	0.5406E+03	Pm-148	0.1659E+06
Ru-103	0.1216E+07	Pu-241	0.1522E+06	Pm-149	0.2481E+06
Ru-105	0.5426E+03	Am-241	0.1897E+03	Pm-151	0.5012E+05
Ru-106	0.5771E+06	Cm-242	0.5649E+05	Pm-148m	0.2899E+05
Rh-105	0.3958E+06	Cm-244	0.1339E+05	Pr-144	0.1015E+07
Sb-127	0.6450E+05	I-130	0.2546E+04	Pr-144m	0.1217E+05
Sb-129	0.1176E+03	Kr-83m	0.3727E+00	Sm-153	0.2171E+06
Te-127	0.7344E+05	Xe-138	0.0000E+00	Y-94	0.0000E+00
Te-127m	0.1222E+05	Xe-131m	0.8276E+04	Y-95	0.0000E+00
Te-129	0.2383E+05	Xe-133m	0.3403E+05	Y-91m	0.1702E+05
Te-129m	0.3637E+05	Xe-135m	0.1434E+04	Br-82	0.2060E+04
Te-131m	0.3690E+05	Cs-138	0.0000E+00	Br-83	0.8833E-01
Te-132	0.6852E+06	Cs-134m	0.5122E+00	Br-84	0.0000E+00
I-131	0.6424E+06	Rb-88	0.4804E+01	Am-242	0.1138E+05
I-132	0.7060E+06	Rb-89	0.0000E+00	Np-238	0.2238E+06
I-133	0.3019E+06	Sb-124	0.1663E+04	Pu-243	0.5681E+03
I-134	0.2087E-09	Sb-125	0.1566E+05		

Table 1.7.6-1 Spent Fuel Cask Drop Source Terms*

*Listed source term is for a single assembly.

Nuclide	30 Day Decay (Curies)	90 Day Decay (Curies)
Co-58	0.0000E+00	0.0000E+00
Co-60	0.0000E+00	0.0000E+00
Kr-85	0.6563E+04	0.6493E+04
Kr-85m	0.0000E+00	0.0000E+00
Kr-87	0.0000E+00	0.0000E+00
Kr-88	0.0000E+00	0.0000E+00
Rb-86	0.3205E+03	0.3450E+02
Sr-89	0.5045E+06	0.2213E+06
Sr-90	0.5178E+05	0.5157E+05
Sr-91	0.1431E-16	0.0000E+00
Sr-92	0.0000E+00	0.0000E+00
Y-90	0.5184E+05	0.5159E+05
Y-91	0.6805E+06	0.3344E+06
Y-92	0.0000E+00	0.0000E+00
Y-93	0.3964E-15	0.0000E+00
Zr-95	0.8884E+06	0.4637E+06
Zr-97	0.1774E-06	0.0000E+00
Nb-95	0.1146E+07	0.7766E+06
Mo-99	0.6714E+03	0.1816E-03
Tc-99m	0.6469E+03	0.1749E-03
Ru-103	0.6102E+06	0.2118E+06
Ru-105	0.0000E+00	0.0000E+00
Ru-106	0.2925E+06	0.2611E+06
Rh-105	0.5545E+00	0.3056E-12
Sb-127	0.3350E+03	0.6807E-02
Sb-129	0.0000E+00	0.0000E+00
Te-127	0.8360E+04	0.5490E+04
Te-127m	0.8207E+04	0.5606E+04
Te-129	0.1138E+05	0.3301E+04
Te-129m	0.1748E+05	0.5071E+04
Te-131m	0.5959E-02	0.2118E-16
Te-132	0.1664E+04	0.4757E-02
I-131	0.5355E+05	0.3038E+03
I-132	0.1715E+04	0.4902E-02
I-133	0.5526E-04	0.7976E-25
I-134	0.0000E+00	0.0000E+00

Nuclide	30 Day Decay (Curies)	90 Day Decay (Curies)
I-135	0.0000E+00	0.0000E+00
Xe-133	0.3313E+05	0.1194E+02
Xe-135	0.0000E+00	0.0000E+00
Cs-134	0.1016E+06	0.9612E+05
Cs-136	0.6310E+04	0.2638E+03
Cs-137	0.6483E+05	0.6459E+05
Ba-139	0.0000E+00	0.0000E+00
Ba-140	0.2450E+06	0.9480E+04
La-140	0.2819E+06	0.1091E+05
La-141	0.0000E+00	0.0000E+00
La-142	0.0000E+00	0.0000E+00
Ce-141	0.6297E+06	0.1752E+06
Ce-143	0.3033E+00	0.2220E-13
Ce-144	0.8358E+06	0.7222E+06
Pr-143	0.2630E+06	0.1226E+05
Nd-147	0.7146E+05	0.1663E+04
Np-239	0.2042E+04	0.1581E+02
Pu-238	0.2174E+04	0.2199E+04
Pu-239	0.2968E+03	0.2968E+03
Pu-240	0.2972E+03	0.2972E+03
Pu-241	0.8941E+05	0.8870E+05
Am-241	0.1070E+03	0.1260E+03
Cm-242	0.2428E+05	0.1882E+05
Cm-244	0.2743E+04	0.2726E+04
I-130	0.5151E-13	0.0000E+00
Kr-83m	0.0000E+00	0.0000E+00
Xe-138	0.0000E+00	0.0000E+00
Xe-131m	0.2962E+04	0.1204E+03
Xe-133m	0.5377E+01	0.3038E-07
Xe-135m	0.0000E+00	0.0000E+00
Cs-138	0.0000E+00	0.0000E+00
Cs-134m	0.0000E+00	0.0000E+00
Rb-88	0.0000E+00	0.0000E+00
Rb-89	0.0000E+00	0.0000E+00
Sb-124	0.5599E+03	0.2806E+03
Sb-125	0.8790E+04	0.8446E+04
Sb-126	0.1278E+03	0.4525E+01
Te-131	0.1341E-02	0.4765E-17
Te-133	0.0000E+00	0.0000E+00
Te-134	0.0000E+00	0.0000E+00

Nuclide	30 Day Decay (Curies)	90 Day Decay (Curies)
Te-125m	0.1902E+04	0.1940E+04
Te-133m	0.0000E+00	0.0000E+00
Ba-141	0.0000E+00	0.0000E+00
Ba-137m	0.6132E+05	0.6110E+05
Pd-109	0.1499E-10	0.0000E+00
Rh-106	0.2925E+06	0.2611E+06
Rh-103m	0.5502E+06	0.1908E+06
Tc-101	0.0000E+00	0.0000E+00
Eu-154	0.6312E+04	0.6229E+04
Eu-155	0.4225E+04	0.4129E+04
Eu-156	0.2550E+05	0.1649E+04
La-143	0.0000E+00	0.0000E+00
Nb-97	0.1911E-06	0.0000E+00
Nb-95m	0.6589E+04	0.3441E+04
Pm-147	0.1206E+06	0.1162E+06
Pm-148	0.4604E+04	0.3221E+03
Pm-149	0.3193E+02	0.2179E-06
Pm-151	0.3097E-02	0.1665E-17
Pm-148m	0.1557E+05	0.5685E+04
Pr-144	0.8358E+06	0.7222E+06
Pr-144m	0.1003E+05	0.8666E+04
Sm-153	0.5430E+01	0.2827E-08
Y-94	0.0000E+00	0.0000E+00
Y-95	0.0000E+00	0.0000E+00
Y-91m	0.9094E-17	0.0000E+00
Br-82	0.1919E-02	0.1012E-14
Br-83	0.0000E+00	0.0000E+00
Br-84	0.0000E+00	0.0000E+00
Am-242	0.1237E+02	0.1236E+02
Np-238	0.1195E+02	0.6211E-01
Pu-243	0.2743E-06	0.2743E-06

Table 1.8.1-1 Release-Receptor Combination Parameters for Analysis Events

Release Point	Receptor Point	Release Height (ft)	Release Height (m)	Receptor Height (ft)	Receptor Height (m)	Distance (ft)	Distance (m)	Direction with respect to true North
Closest Containment Point	Normal Control Room Intake "A"	73.8	22.5	73.8	22.5	79.4	24.2	168
Closest Containment Point	Normal Control Room Intake "B"	73.8	22.5	73.8	22.5	70.5	21.5	174
Closest Containment Point	Emergency Control Room Intake	48.9	14.9	48.9	14.9	312	95.1	202
SIRW Tank Vent	Normal Control Room Intake "A"	80.4	24.5	73.8	22.5	34.8	10.6	157
SIRW Tank Vent	Normal Control Room Intake "B"	80.4	24.5	73.8	22.5	25.3	7.7	184
SIRW Tank Vent	Emergency Control Room Intake	80.4	24.5	48.9	14.9	266	81.2	214
Stack Vent	Normal Control Room Intake "A"	192	58.5	73.8	22.5	73.2	22.3	169
Stack Vent	Normal Control Room Intake "B"	192	58.5	73.8	22.5	65.0	19.8	181
Stack Vent	Emergency Control Room Intake	192	58.5	48.9	14.9	320	97.4	209
ADV	Normal Control Room Intake "A"	57.1	17.4	73.8	22.5	66.0	20.1	192
ADV	Normal Control Room Intake "B"	57.1	17.4	73.8	22.5	65.0	19.8	207
ADV	Emergency Control Room Intake	57.1	17.4	48.9	14.9	329	100.4	214
SSRV (East Bank)	Normal Control Room Intake "A"	57.1	17.4	73.8	22.5	57.7	17.6	187
SSRV (West Bank)	Normal Control Room Intake "B"	57.1	17.4	73.8	22.5	55.8	17.0	204

Release Point	Receptor Point	Release Height (ft)	Release Height (m)	Receptor Height (ft)	Receptor Height (m)	Distance (ft)	Distance (m)	Direction with respect to true North
SSRV (East Bank)	Emergency Control Room Intake	57.1	17.4	48.9	14.9	318	96.8	213
Containment Equipment Door	Normal Intake 'A'	66.9	20.4	73.8	22.5	87.9	26.8	152
Containment Equipment Door	Normal Intake 'B'	66.9	20.4	73.8	22.5	75.8	23.1	161
Containment Equipment Door	Emergency Intake	66.9	20.4	73.8	22.5	313	95.4	204
Turbine Building NE Roof Exhauster	Normal Intake 'A'	89.9	27.4	73.8	22.5	68.2	20.8	257
Turbine Building NE Roof Exhauster	Normal Intake 'B'	89.9	27.4	73.8	22.5	82.7	25.2	264
Turbine Building NE Roof Exhauster	Emergency Intake	89.9	27.4	48.9	14.9	325	99.1	227
Turbine Building NW Roof Exhauster	Normal Intake 'A'	89.9	27.4	73.8	22.5	73.2	22.3	266
Turbine Building NW Roof Exhauster	Normal Intake 'B'	89.9	27.4	73.8	22.5	88.6	27.0	271
Turbine Building NW Roof Exhauster	Emergency Intake	89.9	27.4	48.9	14.9	324	98.6	229

Table 1.8.1-2 Onsite Atmospheric Dispersion (*X/Q*) Factors for Analysis Events

Release – Receptor Pair	Release Point	Receptor Point	0-2 hr X/Q	2-8 hr X/Q	8-24 hr X/Q	1-4 days X/Q	4-30 days X/Q
A	Containment Closest Point	Normal Intake 'B'	1.43E-02	1.11E-02	4.13E-03	3.23E-03	2.49E-03
B	Containment Closest Point	Emergency Intake	7.26E-04	6.18E-04	2.47E-04	1.77E-04	1.30E-04
C	SIRWT Vent	Normal Intake 'B'	9.57E-02	7.59E-02	2.87E-02	2.19E-02	1.65E-02
D	SIRWT Vent	Emergency Intake	9.66E-04	7.92E-04	3.13E-04	2.20E-04	1.64E-04
E	Plant Stack	Normal Intake 'B'	6.10E-03	4.32E-03	1.73E-03	1.27E-03	9.79E-04
F	Plant Stack	Emergency Intake	8.32E-04	7.69E-04	2.83E-04	2.15E-04	1.57E-04
G	Closest ADV	Normal Intake 'A'	1.65E-02	1.34E-02	5.40E-03	4.03E-03	2.98E-03
H	Closest ADV	Emergency Intake	7.36E-04	6.42E-04	2.43E-04	1.75E-04	1.28E-04
I	Closest SSRV	Normal Intake 'A'	-	-	-	4.98E-03	3.72E-03
J	Closest SSRV	Normal Intake 'B'	2.11E-02	1.71E-02	6.91E-03	-	-
K	Closest SSRV	Emergency Intake	7.96E-04	6.91E-04	2.60E-04	1.90E-04	1.37E-04
L	Containment Equipment Door	Normal Intake 'B'	1.25E-02	9.83E-03	3.62E-03	2.86E-03	2.28E-03
M	Containment Equipment Door	Emergency Intake	7.32E-04	6.13E-04	2.45E-04	1.75E-04	1.29E-04
N	Turbine Building NE Roof Exhauster	Normal Intake 'A'	1.31E-02	1.13E-02	4.68E-03	2.87E-03	2.36E-03
O	Turbine Building NE Roof Exhauster	Emergency Intake	-	-	2.58E-04	-	-
P	Turbine Building NW Roof Exhauster	Emergency Intake	7.99E-04	6.43E-04	-	1.75E-04	1.32E-04

Table 1.8.1-3 Release-Receptor Point Pairs Assumed for Analysis Events

Event		
LOCA:	Prior to CR Isolation	Following CR Isolation
- Containment Leakage	A	B
- ECCS Leakage	E	F
- SIRWT Backleakage	C	D
FHA		
- Containment Release	L	M
- FHB Release	E	F
Cask Drop	**Cases 1 & 2**	**Case 3**
Filtered Release - Unfiltered Makeup and Inleakage	E	N/A
Unfiltered Release - Unfiltered Makeup and Inleakage	L	L
Filtered Release - Filtered Makeup	F	N/A
Unfiltered Release - Filtered Makeup	M	N/A
MSLB:		
- Break Release	N	O & P
- MSSV/ADV Release	G	H
SGTR	J & G Initial release via SSRVs switching to ADVs	K & H Initial release via SSRVs switching to ADVs
RCCA Ejection:		
- Containment Leakage	A	B
- Secondary Side Release	J & G Initial release via SSRVs switching to ADVs	K & H Initial release via SSRVs switching to ADVs
Small Line Break Outside Containment	F	E

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	***5.39E-04***	6.66E-5
0-8 hours	3.31E-04	3.03E-5
8-24 hours	2.59E-04	2.04E-5
1-4 days	1.53E-04	8.67E-6
4-30 days	7.14E-05	2.54E-6

The above table summarizes the maximum *X/Q* factors for the EAB and LPZ. Note that the 0-2 hour EAB *X/Q* factor was used for the entire event.

Table 2.1-1 Loss of Coolant Accident (LOCA) – Inputs and Assumptions

Input/Assumption	Value
Release Inputs:	
Core Power Level	2703 MW_{th}
Core Average Fuel Burnup	39,300 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Initial PCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 1.7.2-1
Core Fission Product Inventory	Table 1.7.4-1
Containment Leakage Rate 0 to 24 hours after 24 hours	 0.10% (by weight)/day 0.05% (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)	RG 1.183, Sections 3.1, 3.2, and Table 2
<u>ECCS Systems Leakage (from 19 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 Iodine Decontamination Factor No credit taken for dilution or holdup	 39,054 $ft.^3$ 0.053472 ft^3/min Calculated – 0.03 Used for dose determination – 0.10 97% elemental, 3% organic 2 (based on current design basis)

Input/Assumption	**Value**
SIRWT Back-leakage (from 19 minutes to 30 days) Sump Volume ECCS Leakage to SIRWT (2 times allowed value) Flashing Fraction (elemental iodine assumed to be released into tank space based upon partition factor) SIRWT liquid/vapor elemental iodine partition factor Elemental Iodine fraction in SIRWT	 292,143 gallons (minimum valve for ECCS leakage, maximizes sump iodine concentration) 430,708 gallons (maximum value for SIRWT backleakage to be consist with assumption of minimum water level in SIRWT) 4.4 gpm until 2 hours into the event, then 0.05 gpm 0 % based on temperature of fluid reaching SIRWT Table 2.1-7 Table 2.1-6
Initial SIRWT Liquid Inventory (minimum at time of recirculation)	4,144 gallons
Release from SIRWT Vapor Space	Table 2.1-8
Removal Inputs:	
Containment Aerosol/Particulate Natural Deposition (only credited in unsprayed regions)	0.1/hour
Containment Elemental Iodine Wall Deposition	2.3/hour
Containment Spray Coverage	≥90%
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Aerosol Time to reach DF of 50	 4.8/hour 2.515 hours 1.8/hour (reduced to 0.18 at 3.385 hours) 3.385 hours
Spray Initiation Time	60 seconds (0.016667 hours)
Control Room Ventilation System Time of automatic control room isolation and switch to emergency mode Control Room Unfiltered Inleakage	Table 1.6.3-1 90 seconds 10 cfm
Transport Inputs:	
Containment Leakage Release	Containment closest point
ECCS Leakage	Plant stack

Input/Assumption	Value
SIRWT Backleakage	SIRWT vent
Personnel Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Table 1.8.1-2 and Table 1.8.1-3
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 2.1-2 LOCA Release Phases

Phase	Onset	Duration
Gap Release	30 seconds	0.5 hours
Early In-Vessel	0.5 hours	1.3 hours

* From RG 1.183, Table 4

Table 2.1-3 Time Dependent SIRWT pH

Time (hours)	SIRWT pH
0.00	4.500
0.3167	4.500
0.50	4.505
1.00	4.518
2.00	4.544
2.00	4.544
4.00	4.545
8.00	4.546
16.00	4.548
24.00	4.550
48.00	4.557
72.00	4.563
96.00	4.570
120.00	4.576
144.00	4.583
168.00	4.589
192.00	4.595
240.00	4.607
288.00	4.618
336.00	4.630
384.00	4.641
432.00	4.651
528.00	4.672
624.00	4.692
720.00	4.711

Table 2.1-4 Time Dependent SIRWT Total Iodine Concentration*

Time (hours)	SIRWT Iodine Concentration (gm-atom/liter)
0.00	0.00E+00
0.3167	0.00E+00
0.50	5.77E-07
1.00	2.09E-06
2.00	4.84E-06
2.00	4.84E-06
4.00	4.90E-06
8.00	5.02E-06
16.00	5.25E-06
24.00	5.48E-06
48.00	6.16E-06
72.00	6.82E-06
96.00	7.46E-06
120.00	8.08E-06
144.00	8.68E-06
168.00	9.26E-06
192.00	9.83E-06
240.00	1.09E-05
288.00	1.20E-05
336.00	1.29E-05
384.00	1.39E-05
432.00	1.48E-05
528.00	1.64E-05
624.00	1.79E-05
720.00	1.93E-05

*Includes radioactive and stable iodine isotopes

Table 2.1-5 Time Dependent SIRWT Liquid Temperature

Time (hr)	Temperature (°F)
0.00	100.0
0.3167	100.0
0.50	100.0
1.00	100.0
2.00	100.0
2.00	100.0
4.00	100.5
8.00	101.3
16.00	102.4
24.00	103.2
48.00	104.7
72.00	105.0
96.00	105.0
120.00	104.9
144.00	104.8
168.00	104.8
192.00	104.7
240.00	104.6
288.00	104.6
336.00	104.5
384.00	104.5
432.00	104.5
528.00	104.4
624.00	104.4
720.00	104.4

Table 2.1-6 Time Dependent SIRWT Elemental Iodine Fraction

Time (hr)	Elemental Iodine Fraction
0.00	0.00E+00
0.3167	0.00E+00
0.50	1.25E-02
1.00	4.07E-02
2.00	7.95E-02
2.00	7.95E-02
4.00	8.02E-02
8.00	8.16E-02
16.00	8.42E-02
24.00	8.68E-02
48.00	9.38E-02
72.00	1.00E-01
96.00	1.06E-01
120.00	1.11E-01
144.00	1.15E-01
168.00	1.19E-01
192.00	1.23E-01
240.00	1.29E-01
288.00	1.34E-01
336.00	1.38E-01
384.00	1.41E-01
432.00	1.44E-01
528.00	1.47E-01
624.00	1.49E-01
720.00	1.49E-01

Table 2.1-7 Time Dependent SIRWT Partition Coefficient

Time (hr)	Elemental Iodine Partition Coefficient
0.00	45.65
0.3167	45.65
0.50	45.65
1.00	45.65
2.00	45.65
2.00	45.65
4.00	45.21
8.00	44.53
16.00	43.61
24.00	42.95
48.00	41.74
72.00	41.50
96.00	41.50
120.00	41.58
144.00	41.66
168.00	41.66
192.00	41.74
240.00	41.82
288.00	41.82
336.00	41.89
384.00	41.89
432.00	41.89
528.00	41.97
624.00	41.97
720.00	41.97

Table 2.1-8 Adjusted Release Rate from SIRWT

Time (hours)	Adjusted Iodine Release Rate (cfm)
0.3167	5.7048E-04
2.00	1.1955E-05
8.00	1.2895E-05
24.00	1.4921E-05
72.00	1.7737E-05
168.00	1.9907E-05
240.00	2.1376E-05
336.00	2.2501E-05
432.00	2.3366E-05
624.00	2.3737E-05

Table 2.1-9 LOCA Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
LOCA	12.76	3.27	4.01
Acceptance Criteria	25	25	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 10 cfm

Table 2.2-1 Fuel Handling Accident (FHA) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level Before Shutdown	2703 MW_{th}
Core Average Fuel Burnup	39,300 MWD/MTU
Discharged Fuel Assembly Burnup	39,300 – 58,900 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Maximum Radial Peaking Factor	2.04
Number of Fuel Assemblies Damaged	1 fuel assembly
Delay Before Spent Fuel Movement	48 hours
FHA Source Term for a Single Assembly	Table 1.7.5-1
Water Level Above Damaged Fuel Assembly	22.5 feet minimum
Iodine Decontamination Factors	Elemental – 252 Organic – 1 Overall – 183.07
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 Table 1.8.1-2 Table 1.8.1-3
Control Room Ventilation System Time of Control Room Ventilation System Isolation Control Room Unfiltered Inleakage	 20 minutes 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
FHB Ventilation Filter Efficiencies	Elemental iodine – 94% Organic iodine – 94% Noble gas – n/a

Table 2.2-2 Fuel Handling Accident Dose Consequences

One Fuel Assembly Damaged

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
Elemental iodine DF=285			
FHA in Containment	2.20	0.28	4.04
FHA in FHB With 10% of release via FHB filtered ventilation	2.02	0.25	3.68
FHA in FHB With 34% of release via FHB filtered ventilation	1.60	0.20	2.81
FHA in FHB With 50% of release via FHB filtered ventilation	1.31	0.17	2.22
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 100 cfm

Table 2.3-1 Main Steam Line Break (MSLB) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2703 MW_{th}
Core Average Burnup	39,300 MWD/MTU
Radial Peaking Factor	2.04
Fuel Damage	2% DNB 0% Fuel Centerline Melt
Steam Generator Tube Leakage Rate	0.3 gpm per SG
Time to establish shutdown cooling and terminate steam release	8 hours
Time for PCS to reach 212°F and terminate SG tube leakage	12 hours
PCS Mass	432,977 lb_m
SG Secondary Side Mass	Maximum (Hot Zero Power) – 210,759 lb_m (used for faulted SG to maximize release) Minimum (Hot Full Power) – 141,065 lb_m (used for intact SG to maximize concentration)
Release from Faulted SG	Instantaneous
Steam Release from Intact SGs	Table 2.3-2
Secondary Coolant Iodine Activity prior to accident	0.1 µCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG – none Intact SGs – 100
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Table 1.8.1-2 and Table 1.8.1-3
Control Room Ventilation System Time of manual control room normal intake isolation and switch to emergency mode Control Room Unfiltered Inleakage	Table 1.6.3-1 20 minutes 10 cfm
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factors	RG 1.183 Section 4.2.6

Table 2.3-2 Intact SG Steam Release Rate

Time (hours)	**Intact SG Steam Release (lb_m)**
0 – 8	800,000
8 – 720.0	0

Table 2.3-3 MSLB Dose Consequences

Case	**EAB Dose (1) (rem TEDE)**	**LPZ Dose (2) (rem TEDE)**	**Control Room Dose (2) (rem TEDE)**
MSLB	2.46	0.77	4.98
Acceptance Criteria	25 (3)	25 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 10 cfm
(3) RG 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	2703 MW_{th}
Initial PCS Equilibrium Activity (1.0 µCi/gm DE I-131 and 100/E-bar gross activity)	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	40 µCi/gm DE I-131
Maximum equilibrium iodine concentration	1.0 µCi/gm DE I-131
Duration of accident-initiated spike	8 hours
Steam Generator Tube Leakage Rate	0.3 gpm per SG
Time to establish shutdown cooling and terminate steam release	8 hours
PCS Mass	529,706 lb_m for pre-accident iodine spike case 459,445 lb_m for concurrent iodine spike case
SG Secondary Side Mass	141,065 lb_m per SG (minimum mass used to maximize concentration)
Integrated Mass Release	Table 2.4-2
Secondary Coolant Iodine Activity prior to accident	0.1 µCi/gm DE I-131
Steam Generator Secondary Side Partition Coefficients	Faulted SG (flashed tube flow) – Table 2.4-7 Faulted SG (non-flashed tube flow) – 100 Intact SG – 100
Break Flow Flash Fraction	Table 2.4-3
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Table 1.8.1-2 and Table 1.8.1-3
Control Room Ventilation System Time of manual control room normal intake isolation and switch to emergency mode Control Room Unfiltered Inleakage	20 minutes 100 cfm
Breathing Rates Offsite Control Room	 RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 2.4-2 SGTR Integrated Mass Releases [1]

Time (hours)	Break Flow in Ruptured SG (lb_m)	Steam Release from Ruptured SG (lb_m)	Steam Release from Unaffected SG (lb_m)
0 – 0.196417	24,011.15	0	0
0.196417 - 0.5	37,111.85	44,654	53,574
0.5 - 1.388889	81,281	22,152.3	109,629.6
1.388889 - 2	40,798	15,229.7	75,370.4
2 - 3.638889	64,773	75,485.6	145,983.5
3.638889 - 8	357,126	200,868.4	388,464.5
8 - 720	0	0	0

[1] Flowrate assumed to be constant within time period

Table 2.4-3 SGTR Flashing Fraction for Flow From Broken Tube

Time (seconds)	Flashing Fraction
0	0.110
707.1	0.065
736	0.031
859	0.023
1090	0.006
1800	0.006

Table 2.4-4 40 μCi/gm D.E. I-131 Activities

Isotope	**Activity (μCi/gm)**
Iodine-131	33.2194
Iodine-132	7.6660
Iodine-133	34.4971
Iodine-134	3.0025
Iodine-135	14.6932

Table 2.4-5 Iodine Equilibrium Appearance Assumptions

Input Assumption	**Value**
Maximum Letdown Flow	40 gpm
Assumed Letdown Flow *	44 gpm at 120°F, 2060 psia
Maximum Identified PCS Leakage	10 gpm
Maximum Unidentified PCS Leakage	1 gpm
PCS Mass	459,445 lb_m
I-131 Decay Constant	5.986968E-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}

* maximum letdown flow plus 10% uncertainty

Table 2.4-6 Concurrent (335 x) Iodine Spike Appearance Rate

Isotope	**Appearance Rate (Ci/min)**	**Time of Depletion (hours)**
Iodine-131	58.0966961	> 8
Iodine-132	79.8319317	> 8
Iodine-133	90.1310904	> 8
Iodine-134	74.0318685	> 8
Iodine-135	68.9790622	> 8

Table 2.4-7 Affected Steam Generator Water Level and Decontamination Factors for Flashed Flow

Time (seconds)	Water Level Above U-Tubes (feet)	Calculated Decontamination Factor	Decontamination Factor Used in Analysis
0	0.0 (assumed)*	1.0	1.0
707.1	0.0 (assumed)*	1.0	1.0
736	0.11	1.002299	1.002299
859	0.55	1.045037	1.045037
1090	1.39	1.452436	1.452436
1800	3.97	1.467378	1.467378
5000	6.79	60.03443	1.467378
7200	9.43	38.01867	1.467378
13100	12.34	553073.5	58.16008
28800	15.16	58.16008	58.16008

*It is conservatively assumed that no scrubbing occurs until after the reactor trip at 707.1 seconds. Since the U-tubes remain covered throughout the event, it is also conservatively assumed that at the time of trip the water level is just above the top of the U-tubes. The time-dependent water level after the trip is a function of the allowable primary to secondary leakage, broken tube flow, and MSSV/ADV releases from the affected steam generator. To minimize the water level available for scrubbing, the location of the tube break is assumed to be at the top of the U-tubes.

Table 2.4-8 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	0.99	0.22	3.79
Acceptance Criteria (pre-accident iodine spike)	25 (3)	25 (3)	5 (4)
SGTR concurrent iodine spike	1.17	0.21	3.48
Acceptance Criteria (concurrent iodine spike)	2.5 (3)	2.5 (3)	5 (4)

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 100 cfm
(3) RG 1.183, Table 6
(4) 10CFR50.67

Table 2.5-1 Small Line Break Outside of Containment – Inputs and Assumptions

Input/Assumption	Value
PCS Equilibrium Activity	Table 1.7.2-1
Break Flow Rate	160 gpm
Break Temperature	135°F
Break Pressure	35 psia
Time required to isolate break	60 minutes
Maximum equilibrium iodine concentration	1.0 μCi/gm DE I-131
Iodine appearance rate for concurrent iodine spike (500x)	Table 2.5-2
Iodine fraction released from break flow	10%
Reactor building ventilation system filtration	None
Atmospheric Dispersion Factors Offsite Onsite	 Table 1.8.2-1 Table 1.8.1-2 and Table 1.8.1-3
Control Room Ventilation System Time of manual control room normal intake isolation and switch to emergency mode Control Room Unfiltered Inleakage	 20 minutes 100 cfm
Breathing Rates Offsite Onsite	 RG 1.183 Section 4.1.3 RG 1.183 Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 2.5-2 Concurrent (500 x) Iodine Spike Appearance Rate

Isotope	Appearance Rate (Ci/min)
Iodine-131	86.7114868
Iodine-132	119.152137
Iodine-133	134.524016
Iodine-134	110.495326
Iodine-135	102.953824

Table 2.5-3 Small Line Break Outside of Containment Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Small Line Break	0.41	0.05	0.53
Acceptance Criteria	2.5	2.5	5.0

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 100 cfm

Table 2.6-1 Control Rod Ejection – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	2703 MW_{th}
Core Average Fuel Burnup	39,300 MWD/MTU
Fuel Enrichment	3.0 – 5.0 w/o
Maximum Radial Peaking Factor	2.04
% DNB Fuel	14.7%
% Fuel Centerline Melt	0.5%
LOCA Source Term	Table 1.7.4-1
Initial PCS Equilibrium Activity (1.0 μCi/gm DE I-131 and 100/E-bar gross activity)	Table 1.7.2-1
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 RG 1.183
Release From Fuel Centerline Melt Fuel	Section 1 of Appendix H to RG 1.183
Steam Generator Secondary Side Partition Coefficient	100
Steam Generator Tube Leakage	0.3 gpm per SG
Time to establish shutdown cooling	8 hours
PCS Mass	432,976.8 lb_m
SG Secondary Side Mass	minimum – 141,065 lb_m (per SG) Minimum mass used for SGs to maximize steam release nuclide concentration.
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 Table 1.8.1-2 and Table 1.8.1-3
Control Room Ventilation System Time of Control Room Ventilation System Isolation Control Room Unfiltered Inleakage	 20 minutes 10 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
Containment Volume Containment Leakage Rate 0 to 24 hours after 24 hours	1.64E+06 ft^3 0.10% (by weight)/day 0.05% (by weight)/day
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr^{-1} Elemental Iodine – 1.3 hr^{-1} (1) Organic Iodine – None

(1) Conservatively assumes lower elemental iodine natural deposition coefficient than LOCA

Table 2.6-2 Control Rod Ejection Steam Release

Time	SG Steam Release (lb_m)
0 – 1100 sec	107,158.8
1100 sec – 0.5 hours	31,336.8
0.5 hr – 8 hr	1,007,100
>8 hr	0

Table 2.6-3 Control Rod Ejection Dose Consequences

Case	EAB Dose(1) (rem TEDE)	LPZ Dose(2) (rem TEDE)	Control Room Dose(2) (rem TEDE)
CEA Ejection – Containment Release	2.70	0.43	1.14
CEA Ejection – Secondary Release	2.61	0.68	1.14
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 10 cfm

Table 2.7-1 Spent Fuel Cask Drop– Inputs and Assumptions

Input/Assumption	Value
Core Power Level Before Shutdown	2703 MW_{th}
Core Average Fuel Burnup	39,300 MWD/MTU
Discharged Fuel Assembly Burnup	39,300 – 58,900 MWD/MTU
Fuel Enrichment	3.0 – 5 w/o
Number of Fuel Assemblies Damaged	73
Delay Before Cask Drop	Cases 1 & 2 – 30 days Case 3 – 90 days
Source Terms	Table 1.7.6-1
Water Level Above Damaged Fuel Assembly	23.4 feet minimum
Iodine Decontamination Factors	Elemental – 285 Organic – 1 Overall - 200
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Atmospheric Dispersion Factors	Table 1.8.1-2 and Table 1.8.1-3
Control Room Ventilation System Time of Control Room Ventilation System Isolation Control Room Unfiltered Inleakage	 Cases 1 & 2 – 0 seconds Case 3 - ∞ 100 cfm
Time of Control Room Filtered Makeup Flow	Cases 1 & 2 – 0 seconds Case 3 - ∞
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 2.7-2 Spent Fuel Cask Drop Dose Consequences

Case	EAB Dose (1) (rem TEDE)	LPZ Dose (2) (rem TEDE)	Control Room Dose (2) (rem TEDE)
Cask Drop - Case 1	2.04	0.25	1.37
Cask Drop - Case 2	2.78	0.35	1.99
Cask Drop – Case 3	0.08	0.01	1.67
Acceptance Criteria	6.3	6.3	5

(1) Worst 2-hour dose
(2) Integrated 30-day dose based on an unfiltered Control Room inleakage rate of 100 cfm

Table 3-1 Palisades Summary of Alternative Source Term Analysis Results

Case	Unfiltered CR Inleakage (cfm)	EAB Dose[1] (rem TEDE)	LPZ Dose[2] (rem TEDE)	Control Room Dose[2] (rem TEDE)
LOCA	10	12.76	3.27	4.01
MSLB	10	2.46	0.77	4.98
SGTR Pre-accident Iodine Spike	100	0.99	0.22	3.79
Acceptance Criteria		**≤ 25**[3]	**≤ 25**[3]	**≤ 5**[4]
SGTR Concurrent Iodine Spike	100	1.17	0.21	3.48
Small Line Break Outside of Containment	100	0.41	0.05	0.53
Acceptance Criteria		**≤ 2.5** [3]	**≤ 2.5** [3]	**≤ 5** [4]
FHA in Containment	100	2.20	0.28	4.04
FHA in FHB 10% Release Filtration	100	2.02	0.25	3.68
FHA in FHB 34% Release Filtration	100	1.60	0.20	2.81
FHA in FHB 50% Release Filtration	100	1.31	0.17	2.22
Control Rod Ejection – Containment Release	10	2.70	0.43	1.14
Control Rod Ejection – Secondary Release	10	2.61	0.68	1.14
Spent Fuel Cask Drop 30 days Decay 90% of Release via FHB filtration system	100	2.04	0.25	1.37
Spent Fuel Cask Drop 30 days Decay 82.5% of Release via FHB filtration system	100	2.78	0.35	1.99
Spent Fuel Cask Drop 90 days Decay No Control Room Isolation	100	0.08	0.01	1.67
Acceptance Criteria		**≤ 6.3**[3]	**≤ 6.3**[3]	**≤ 5**[4]

(1) Worst 2-hour dose
(2) Integrated 30-day dose
(3) RG 1.183, Table 6
(4) 10CFR50.67