CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

SERIAL: HNP-01-107
10CFR50.4
10CFR50.30(b)
10CFR50.67

JUL 1 7 2001

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
APPLICATION OF AN ALTERNATE SOURCE TERM
METHODOLOGY IN SUPPORT OF
THE STEAM GENERATOR REPLACEMENT
AND POWER UPRATE LICENSE AMENDMENT APPLICATIONS

Dear Sir or Madam:

By letters dated October 4, 2000 and December 14, 2000, Carolina Power & Light Company (CP&L) submitted license amendment requests to revise the Harris Nuclear Plant (HNP) Facility Operating License and Technical Specifications to support steam generator replacement and to allow operation at an uprated reactor core power level of 2900 megawatts thermal (Mwt). CP&L proposes to revise the analyses of radiological consequences previously provided by the October 4, 2000 and December 14, 2000 submittals.

Enclosure 1 provides a description of the proposed changes and the basis for the changes. Enclosure 2 provides, in accordance with 10CFR50.91(a), the basis for CP&L's determination that the proposed changes to our aforementioned amendment requests does not involve a significant hazards consideration. Enclosure 3 to this letter provides an environmental evaluation that demonstrates the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental assessment is required for approval of the enclosed changes to our aforemention amendment requests.

A001

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Enclosure 4 to this letter is a change to a previously proposed change to TS page 3/4 4-29. A proposed change to this same TS page was submitted within Enclosure 5 of our October 4, 2000 license amendment request. The mark-up to TS page 3/4 4-29 provided herein effectively withdraws our previously proposed change to the limitation on Reactor Coolant Dose Equivalent I-131 of 1 micro Curie per gram. Therefore, please replace the previously submitted mark-up to TS page 3/4 4-29 with the revised mark-up page provided herein.

Enclosure 5 provides the summary reports of the Alternate Source Term Analyses. These summary reports are being provided as replacement sections to the corresponding report sections provided in the October 4, 2000 license amendment request (ref.: HNP-00-142, dated October 4, 2000). Enclosed Report Section 2.22 replaces the Report Section 2.22 provided in Enclosure 7 of our October 4, 2000 submittal, and enclosed Report Section 6.3.3 replaces the Report Section 6.3.3 provided in Enclosure 6 of our October 4, 2000 submittal. The revised report sections have been prepared as "one-for-one" replacements to the existing Report Sections to facilitate the staff's replacement and review of this material.

CP&L requests staff review of the enclosed information in conjunction with the staff's ongoing review of the October 4, 2000 and December 14, 2000 license amendment requests for steam generator replacement and power uprate, respectively. NRC issuance of the requested amendments is requested to support HNP refueling outage 10, scheduled to begin on September 22, 2001, and to allow for implementation within 60 days of issuance to allow adequate time for procedure revision and orderly incorporation of the Technical Specification changes.

Please refer any questions regarding the enclosed information to Mr. Mark Ellington at (919) 362-2057.

Sincerely,

James Scarola

James Scarola
Vice President
Harris Nuclear Plant

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge, and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

Darlene S. Yarborough
Notary (Seal)

My commission Expires: 2-21-2005

KWS/kws

Enclosures

c: Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, NCDENR
Mr. N. Kalyanam, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

DESCRIPTION OF THE PROPOSED CHANGES

The Harris Nuclear Plant (HNP) licensing basis for the radiological consequences analyses for Chapter 15 of the FSAR is currently based on methodologies and assumptions that are derived from TID-14844 and other early guidance.

As documented in a draft NEI 99-03, dated January 2000, several nuclear plants performed testing on control room unfiltered inleakage that demonstrated leakage rates in excess of amounts assumed in the accident analyses.

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. The alternative source term methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for HNP to support the increase of the control room unfiltered inleakage.

The following FSAR Chapter 15 accidents are analyzed:

Large Break Loss of Coolant Accident (LBLOCA)
Steam Generator Tube Rupture (SGTR)
Locked Rotor
Single Rod Control Cluster Control Assembly (RCCA) Withdrawal
Loss of Offsite Power (LOOP)
Rod Ejection
Small Break LOCA
Main Steamline Break (MSLB)
Fuel Handling Accident (FHA)
Letdown Line Break
Waste Gas Decay Tank (WGDT) Rupture

Each accident and the specific input assumptions are described in detail in the enclosed reports. These analyses provide for a control room unfiltered inleakage of 300 cfm. The use of 300 cfm unfiltered inleakage as a design basis value is expected to be well above the unfiltered inleakage value determined through testing or analysis consistent with resolution of issues identified in NEI 99-03.

Dose Consequence Results

The dose consequence results with respect to the site boundary [or exclusion area boundary (EAB)], low population zone (LPZ), and the control room are provided below for each of the analyzed FSAR Chapter 15 accidents.

Large Break Loss of Coolant Accident Doses

Exclusion Area Boundary	7.53 rem TEDE
Low Population Zone	4.33 rem TEDE
Control Room	4.99 rem TEDE

The offsite dose limit for a LOCA is 25 rem TEDE per RG 1.183. This is the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Steam Generator Tube Rupture Accident Doses

The SGTR accident doses are listed below.

For the pre-accident iodine spike:

Exclusion Area Boundary	2.20 rem TEDE
Low Population Zone	0.60 rem TEDE
Control Room	1.60 rem TEDE

For the accident-initiated iodine spike:

Exclusion Area Boundary	1.30 rem TEDE
Low Population Zone	0.40 rem TEDE
Control Room	0.90 rem TEDE

The doses at the exclusion area boundary (EAB) and the low population zone (LPZ) for an SGTR with an assumed pre-accident iodine spike must be within the RG 1.183 limit of 5 rem TEDE. The doses at the EAB and the LPZ for an SGTR with an assumed accident-initiated iodine spike must be within the RG 1.183 limit of 2.5 rem TEDE. The doses in the control room must be less than the 10CFR50.67 dose limit of 5 rem TEDE.

Locked Rotor Accident Doses

The locked rotor doses are:

Exclusion Area Boundary	1.89 rem TEDE
Low Population Zone	1.40 rem TEDE
Control Room	3.17 rem TEDE

The offsite dose limit for a locked rotor is 2.5 rem TEDE per RG 1.183. This is 10% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Single RCCA Withdrawal Accident Doses

The single RCCA withdrawal doses are:

Exclusion Area Boundary	1.57 rem TEDE
Low Population Zone	1.23 rem TEDE
Control Room	2.63 rem TEDE

The offsite dose limit for a single RCCA withdrawal accident is not defined in RG 1.183; however, the locked rotor offsite dose limit is 2.5 rem TEDE per RG 1.183. This is 10% of the guideline value of 10CFR50.67. Since the locked rotor event involves similar release mechanisms, its acceptance criteria will be assumed to apply to this single RCCA withdrawal accident. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Main Steam Line Break (MSLB) Doses

The MSLB accident doses are listed below.

For the pre-accident iodine spike:

Site Boundary	0.13 rem TEDE
Low Population Zone	0.14 rem TEDE
Control Room	0.36 rem TEDE

For the accident-initiated iodine spike:

Site Boundary	0.70 rem TEDE
Low Population Zone	1.04 rem TEDE
Control Room	2.47 rem TEDE

For the fuel failure:

Site Boundary	1.44 rem TEDE
Low Population Zone	2.52 rem TEDE
Control Room	3.95 rem TEDE

The offsite dose limit for a MSLB with a pre-accident iodine spike or fuel damage is 25 rem TEDE per RG 1.183. This is the guideline value of 10CFR50.67. For a MSLB with an accident-initiated iodine spike, the offsite dose limit is 2.5 rem TEDE per RG 1.183. This is 10% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Loss of Offsite Power Accident Doses

The loss of offsite power accident doses are listed below:

For the pre-accident iodine spike:

Site Boundary	0.012 rem TEDE
Low Population Zone	0.0092 rem TEDE
Control Room	0.028 rem TEDE

For the accident-initiated iodine spike:

Site Boundary	0.043 rem TEDE
Low Population Zone	0.022 rem TEDE
Control Room	0.065 rem TEDE

The offsite dose limit for the loss of offsite power accident is not defined in RG 1.183. The offsite dose limit for a MSLB with a pre-accident iodine spike or fuel damage is 25 rem TEDE per RG 1.183. This is the guideline value of 10CFR50.67. For a MSLB with an accident-initiated iodine spike, the offsite dose limit is 2.5 rem TEDE per RG 1.183. This is 10% of the guideline value of 10CR50.67. Since the MSLB event involves similar iodine spiking, its acceptance criteria will be assumed to apply to this LOOP accident. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Rod Ejection Accident Doses

The rod ejection doses are:

Exclusion Area Boundary	3.90 rem TEDE
Low Population Zone	4.00 rem TEDE
Control Room	4.30 rem TEDE

The offsite dose limit for a rod ejection is 6.3 rem TEDE per RG 1.183. This is approximately 25% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Small Break Loss of Coolant Accident Doses

The SBLOCA doses are:

Exclusion Area Boundary	9.24 rem TEDE
Low Population Zone	2.83 rem TEDE
Control Room	4.10 rem TEDE

The offsite dose limit for a LOCA is 25 rem TEDE per RG 1.183. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Fuel Handling Accident in Containment Doses

The fuel handling accident in containment doses are:

Site Boundary	2.03 rem TEDE
Low Population Zone	0.46 rem TEDE
Control Room	1.39 rem TEDE

The offsite dose limit for a fuel handling accident is 6.3 rem TEDE per RG 1.183. This is approximately 25% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Fuel Handling Accident In Fuel Handling Building Doses

The fuel handling accident in fuel building doses are:

Site Boundary	0.34 rem TEDE
Low Population Zone	0.077 rem TEDE
Control Room	0.12 rem TEDE

The offsite dose limit for a fuel handling accident is 6.3 rem TEDE per RG 1.183. This is approximately 25% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Letdown Line Break Accident Doses

The small line break outside of containment (SLBOC) doses are:

Exclusion Area Boundary	2.34 rem TEDE
Low Population Zone	0.53 rem TEDE
Control Room	1.50 rem TEDE

The offsite dose acceptance criteria from SRP 15.6.2 is designated as 10% of 10CFR100 limits. Applying this same basis to the 25 rem TEDE in 10CFR50.67, the limit for offsite doses is 2.5 rem TEDE. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

Waste Gas Decay Tank Rupture Accident Doses

The gas decay tank rupture doses are:

Exclusion Area Boundary	0.30 rem TEDE
Low Population Zone	0.069 rem TEDE
Control Room	0.049 rem TEDE

The offsite dose limit for a gas decay tank rupture is defined in HNP Technical Specification 6.8.4j as 0.5 rem whole body. This translates to a dose limit of 0.5 rem TEDE. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67. Therefore, the acceptance criteria are met.

10 CFR 50.92 EVALUATION

The commission has provided standards in 10CFR50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. CP&L has reviewed this proposed license amendment request and determined that its adoption would not involve a significant hazards determination. The bases for this determination are as follows:

Proposed Change

HNP proposes to revise the FSAR Chapter 15 accident analyses to adopt the alternate source term methodology using the guidance of NRC Regulatory Guide 1.183.

Basis

This change does not involve a significant hazards consideration for the following reasons:

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

   An alternative source term calculation has been performed for HNP which demonstrates that dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed change does not modify the design or operation of the plant.

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

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

   The proposed change does not affect plant structures, systems, or components. The operation of plant systems and equipment will not be affected by this proposed change.

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

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

   The proposed change is the implementation of the alternate source term methodology consistent with NRC Regulatory Guide 1.183. The proposed change does not significantly affect any of the parameters that relate to the margin of safety as described in the Bases of the TS or FSAR. Accordingly, NRC Acceptance Limits are not significantly affected by this change.

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

ENVIRONMENTAL CONSIDERATIONS

10 CFR 51.22(c)(9) provides criterion for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite: (3) result in a significant increase in individual or cumulative occupational radiation exposure. CP&L has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Proposed Change

HNP proposes to revise the FSAR Chapter 15 accident analyses to adopt the alternate source term methodology using the guidance of NRC Regulatory Guide 1.183.

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9) for the following reasons:

1. As demonstrated in Enclosure 2, the proposed amendment does not involve a significant hazards consideration.

2. The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite.

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

3. The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure.

   The proposed change is purely analytical and does not result in any physical plant changes or new surveillances that would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no significant affect on either individual or cumulative occupational radiation exposure.

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/OPERATING LICENSE NO. NPF-63

TECHNICAL SPECIFICATION PAGE 3/4 4-29

Note: This TS page (mark-up and retyped page) is being submitted as a replacement page for the proposed changes to this same page submitted by Enclosure 5 to HNP-00-142 (i.e., our October 4, 2000 license amendment request for steam generator replacement)

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/E microCuries per gram of gross radioactivity.

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

ACTION:

Add

60.0 microCurie per gram DOSE EQUIVALENT I-131

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for ~~more than 48 hours during one continuous~~ time interval or exceeding ~~the limit line shown on Figure 3.4-1~~, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. The provisions of Specification 3.0.4 are not applicable. Delete Delete

b. With the specific activity of the reactor coolant greater than 100/E microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

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

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

SURVEILLANCE REQUIREMENTS

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

*With T_{avg} greater than or equal to 500°F.

Add

Amendment No.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

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

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding 60.0 microCurie per gram DOES EQUIVALENT I-131, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the reactor coolant greater than 100/Ē microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

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

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

SURVEILLANCE REQUIREMENTS

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

*With T_{avg} greater than or equal to 500°F.

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/OPERATING LICENSE NO. NPF-63

ALTERNATE SOURCE TERM ACCIDENT ANALYSES

2.22 Personnel Radiation Dose Analysis

The Harris Nuclear Plant (HNP) radiation dose analysis has been conducted to determine the radiological impact of Steam Generator Replacement and Power Uprate Project (SGR/Uprate) which includes operation of the Model Delta 75 replacement steam generators (RSGs) at the uprated reactor core power level of 2900 MWt (NSSS power level of 2912.4 MWt.). Accident Analysis is based on 102% power, or 2958 MWt.

2.22.1 Introduction and Background

The impact of SGR/Uprate on radiation dose analysis at the HNP encompasses the following radiological dose evaluations: 1) Normal Operation Doses to the Public From Gaseous and Liquid Releases, 2) Normal Operation Doses to Onsite Personnel, 3) Design Basis Accident Doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ), and 4) Design Basis Accident Doses To Onsite Personnel in Vital Areas, including the Control Room.

- Normal Operation Doses to the Public from Gaseous and Liquid Releases

Releases to the environment from normal operations after the SGR/Uprate were determined based on NUREG-0017, as implemented by the GALE Code. The offsite dose impacts from the releases were determined by using the computer programs GASPAR and LADTAP. This methodology is consistent with the HNP pre-SGR/Uprate design basis evaluations. Offsite doses continue to meet 10CFR50, Appendix I dose criteria while the effluent concentrations remain within 10CFR20 limitations.

- Normal Operation Doses to Onsite Personnel

The reduction in reactor coolant activity, relative to previously analyzed conditions, serves to reduce in-plant radiation levels and associated personnel doses. The existing radiation zoning remains conservative for the SGR/Uprate operating conditions.

- Design Basis Accident Doses at the EAB and LPZ

All design basis accident EAB and LPZ doses have been re-evaluated to reflect the SGR/Uprate; this includes changes to core source terms and fuel failure assumptions. The methodology in Regulatory Guide (RG) 1.183 was used in evaluating doses at the EAB and LPZ. All calculated offsite doses remain within 10CFR50.67 limits, and within fractions of 10CFR50.67 limits as prescribed in RG 1.183.

- Design Basis Accident Doses to Onsite Personnel in Vital Areas

All design basis accidents have been re-evaluated for the Control Room (CR), Technical Support Center (TSC), and the Emergency Operations Facility (EOF). In addition to the methodology changes discussed in the previous section, an increase in unfiltered in-leakage to 300 cfm has been assumed for the control room to provide additional operational margin. All doses remain within 10CFR50.67 limits as prescribed in RG 1.183.

Accessibility to other Vital Areas is not significantly impacted by the SGR/Uprate changes. Post-Accident dose rates in the Reactor Auxiliary Building (RAB) were re-evaluated. The principal area requiring access is for the Post Accident Sampling System. Dose rates in this area increase between 5 and 8%, due to SGR/Uprate, as a result of consideration of fission product daughters in the Emergency Core Cooling System water. However, accessibility to this and other previously identified facilities is maintained.

Configuration Changes:

The SGR/Uprate does not affect existing radiological protection features. The analysis of the radiation dose impact on plant personnel and the general public does not involve a change to any system function credited in the FSAR for radiation mitigation and protection. The plant layout and shielding, designed to minimize personnel exposure, were not affected.

Revised Process Conditions:

Operating the plant at SGR/Uprate conditions slightly increases the generation of fission in the core and generation of activation products in the Reactor Coolant System (RCS). For accident conditions, bounding fuel failure assumptions lead to increases in accident basis RCS inventories used in many events which release RCS actuals. In addition, changes to the calculation methodology, assumptions, and operational considerations have affected the determination of onsite personnel, general public, and equipment doses for normal and accident conditions.

The results of the radiological consideration for the SGR/Uprate are presented in section 2.22.4.

2.22.2 Description of Analyses and Evaluations

Evaluations of the radiological impact of SGR/Uprate on site personnel and the general public was performed for normal and accident conditions. These evaluations are dependent upon:

- Current radioactivity source terms used in environmental determinations.
- Revised calculational methodology.
- Plant configuration (post SGR/Uprate).
- The physical nature of the sources (e.g., airborne, liquid, contained in piping, and deposited on filters).

A shielding and onsite dose evaluation was conducted, taking into account revised source terms and mass releases resulting from SGR/Uprate. It was determined that SGR/Uprate did not require any changes to HNP shielding. This review was conducted in conjunction with a review of the Reactor Coolant System (RCS) and core inventory.

Assumptions:

- Normal shielding reviews are based on 1% defective fuel. Accident shielding is based on 100% core melt source terms.

- The following inputs were changed relative to the normal offsite release and dose assumptions described in Final Safety Analysis (FSAR) Chapter 11: 1) the increase in letdown flow, 2) changes in secondary system blowdown and condensate system operations, 3) only new 10CFR20 effluent concentration (EC) limits are addressed, and 4) filtration of fuel handling building normal releases was previously erroneously credited, this has been corrected.

- The SGR/Uprate core source terms are based on a general parametric analysis of 18 month fuel cycle conditions. The analysis was performed using the ORIGEN computer code. This program is identified in the FSAR as a basis for source terms. A single set of enveloping source terms was generated, so that separate accident dose assessments are no longer required. These bounding source terms are used for onsite and offsite dose assessment.

- Fuel failure assumptions for several non-LOCA design basis accidents have been selected to envelop those which might be calculated in future reloads. Due to this set of higher, more bounding, fuel damage assumptions, the offsite doses for the following accidents, in some cases, may now be closer to their limits: 1) Main Steam Line Break, 2) Locked Rotor, 3) Single Rod Cluster Control Assembly (RCCA) Withdrawal, 4) Misloaded Core, 5) RCCA Rejection.

- New 10CFR20 EC limits are addressed.

- RG 1.183 methodology was used.

- Iodine spiking model as described in the appropriate sections of RG 1.183, are included in those FSAR accidents where applicable. Although not specifically required by RG 1.183, a pre-existing iodine spike is modeled for the loss of offsite power event to provide a consistent treatment for all of the analyses.

Operational Changes:

- Increased Letdown Flow:

  The revised Chemical and Volume Control System operation, in support of SGR/Uprate, will increase the letdown flow to include one 45 gpm orifice and one 60 gpm orifice instead of just one 60 gpm orifice as previously analyzed. The PWR-GALE computer code was used to calculate the release of radioactive material given a letdown flow rate of 106 gpm. The result of this change combined with the increase in reactor coolant volume is a decrease in normal operation reactor coolant inventory, directly resulting in a decrease in normal offsite doses. It also assures that spent fuel pool cooling water activity will remain within analyzed concentrations.

- Steam Generator (SG) Blowdown Processing vs. Condensate Polisher Use:

Previous analysis assumes no treatment of the SG blowdown before it is sent to the main condenser, for cleanup by the condensate polisher. In the revised analysis the SG blowdown is demineralized, prior to being sent to the condenser. The analysis assumes that condensate polishing is not required except for chemistry control during startup. In the event of indications of primary to secondary coolant leakage, condensate resins are not regenerated, so this liquid release path will be eliminated.

2.22.3 Acceptance Criteria

Shielding for normal operations must meet the requirements of 10CFR20 related to operator dose and access control. Additional guidance for shielding is provided by USNRC Regulatory Guide 8.8 as described in FSAR Sections 12.1 and 12.3. The design of radwaste equipment must be such that the plant is capable of maintaining offsite releases and resulting doses within the requirements of 10CFR20 and 10CFR50, Appendix I. Additional guidance for evaluating compliance with these requirements is taken from USNRC Regulatory Guides 1.109 through 1.113, as discussed in FSAR Sections 11.2.3 (liquid) and 11.3.3 (gaseous). Actual performance and operation of installed equipment and reporting of actual offsite releases and doses continues to be controlled by the requirements of the Offsite Dose Calculation Manual (ODCM).

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 by the NRC in Table 6 of RG 1.183.

Input assumption guidance for specific accidents is taken from USNRC Regulatory Guide 1.183 (refer to FSAR Section 1.8 for CP&L's compliance to each RG).

1.52 ESF Filter Systems
1.78 Control Room Habitability
1.109 Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents
1.111 Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors
1.112 Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors
1.113 Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I
1.143 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

2.22.4 Results

The plant will continue to satisfy required radiation protection requirements following the SGR/Uprate.

The results of the evaluation are broken down into the following subsections:

2.22.4.1 Shielding

The SGR/Uprate does not change system and component functions described in the FSAR. The plant layout and shielding, designed to minimize personnel exposure, were not affected. The original shielding design was based on radiation source terms developed from a reactor core thermal power of 2900 MWt (NSSS power level of 2912.4 MWt) and the equivalent of 1 percent fuel cladding defects. For SGR/Uprate, the RCS, core, and waste gas activities are based on 102 percent of the uprate core power of 2900 MWt. Significant conservatism was included in the originally calculated dose rate for shielding design. As a result, the increase in dose rate due to SGR/Uprate does not create additional inaccessible areas in the plant. Additionally, increased reactor coolant letdown processing is expected to reduce in-plant doses.

2.22.4.2 Normal Offsite Releases and Doses

The original bounding calculations prepared to evaluate conformance to 10CFR20 and 10CFR50, Appendix I demonstrate that sufficient radwaste equipment is provided in the HNP design to maintain releases within the limits of 10CFR20, Appendix B, and the resulting offsite dose to the most exposed individual within the limits of 10CFR50, Appendix I. No hardware modifications to the radwaste system are required to support SGR/Uprate.

2.22.4.3 Accident Doses

Introduction

The Shearon Harris Nuclear Power Plant (HNP) licensing basis for the radiological consequences analyses for Chapter 15 of the FSAR is currently based on methodologies and assumptions that are derived from TID-14844 (Reference 1) and other early guidance.

Regulatory Guide (RG) 1.183 (Reference 2) 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 (Reference 3). The alternative source term methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for the HNP to support the increase of the control room unfiltered inleakage. The following FSAR chapter 15 accidents are analyzed: Large Break Loss of Coolant Accident (LBLOCA), Steam Generator Tube Rupture (SGTR), Locked Rotor, Single Rod Control Cluster Assembly (RCCA) Withdrawal, Loss of Offsite Power, Rod Ejection, Small Break Loss of Coolant Accident (SBLOCA), Main Steamline Break (MSLB), Fuel Handling Accident (FHA), Letdown Line Break, and Waste Gas Decay Tank (WGDT) Rupture. Each accident and the specific input assumptions are described in detail in subsequent sections in this report.

Common Analysis Inputs and Assumptions

The assumptions and inputs described in this section are common to analyses discussed in this report. The accident specific inputs and assumptions are discussed in Sections 2.22.4.3.1 through 2.22.4.3.12.

The total effective dose equivalent (TEDE) doses are determined at the exclusion area boundary (EAB) for the worst 2-hour interval. The TEDE doses at the low population zone (LPZ) and for the control room personnel (CR) are determined for the duration of the event. The interval for determining control room doses may extend beyond the time when the releases are terminated. This accounts for the additional dose to the operators in the control room, which will continue for as long as the activity is circulating within the control room envelope.

The TEDE dose is equivalent to the committed effective dose equivalent (CEDE) or inhalation dose plus the acute dose (EDE) dose for the duration of exposure to the cloud. The dose conversion factors (DCFs) used in determining the CEDE dose are from Reference 5 and are given in Table 2.22-1. The dose conversion factors used in determining the EDE dose are from Reference 10 and are listed in Table 2.22-2.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the offsite radiological calculations are provided in Table 2.22-3.

Parameters used in the control room personnel dose calculations are provided in Table 2.22-4. These parameters include the normal operation flowrates, the emergency operation flowrates, control room volume, filter efficiencies and control room operator breathing rates. The atmospheric dispersion factors are used to determine the activity available at the intake. These factors bound credible release points for all events, and are the same factors used in the current HNP control room dose calculations except for those used for the ECCS leakage to the RWST. For the ECCS leakage to the RWST, these existing factors were used to develop a specific, conservative set of atmospheric dispersion factors for this particular release point/receptor pair. The inflow (filtered and unfiltered) to the control room and the control room recirculation flow are used to calculate the activity introduced to the control room and cleanup of activity from that flow.

The core fission product activity is provided in Table 2.22-7 for all nuclides that are addressed. The Technical Specification nominal reactor coolant activity based on 1% fuel defects for noble gases and other nuclides and 1.0 μCi/gm Dose Equivalent (DE) I-131 for iodine nuclides is provided in Table 2.22-8. The core and coolant activities in Tables 2.22-7 and 2.22-8 are based on a core power of 2900 MWt increased to 2958 to cover 2% uncertainty. Decay constants for each nuclide are provided in Table 2.22-9.

2.22.4.3.1 Large Break Loss of Coolant Accident Doses

An abrupt failure of the main reactor coolant pipe is assumed to occur and it is assumed that the emergency core cooling features fail 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 from the core is released to the containment and from there released to the environment by means of containment leakage and leakage from the emergency core cooling system.

Comparison of NUREG-1465 Source Term Methodology to TID-14844

The reanalysis of the LBLOCA offsite and control room doses for HNP uses the following RG 1.183 source term characteristics in place of those identified in TID-14844 and RG 1.4:

Iodine chemical species
Fission product release timing
Fission product release fractions
Fission product groups

A comparison of RG 1.183 to the model defined in TID-14844 and RG 1.4 is provided in Tables 2.22-10 through 2.22-12.

Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 2.22-14. Activity is released from the fuel into the containment using the timing and release fractions from Tables 2.22-11 and 2.22-12. The analysis considers the release of activity from the containment via containment leakage. In addition, once the recirculation mode of the emergency core cooling system (ECCS) is established, activity in the sump solution may be released to the environment by means of leakage from ECCS equipment into the auxiliary building. Activity of the sump solution may also be released to the environment by means of leakage into the refueling water storage tank (RWST). The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths.

The following sections address topics of significant interest.

Source Term

The reactor coolant activity is assumed to be released over the first 30 seconds of the accident. However, the activity in the coolant is insignificant compared with the release from the core and is not included in the analysis.

The use of RG 1.183 source term modeling results in several major departures from the assumptions used in the existing LOCA dose analysis as reported in the FSAR:

- Instead of assuming instantaneous melting of the core and release of activity to the containment, the release of activity from the core occurs over a 1.8 hour interval.

- Instead of considering only the release of iodines and noble gases, a wide spectrum of nuclides is taken into consideration. Table 2.22-13 lists the nuclides being considered for the LOCA with core melt (eight groups of nuclides). Tables 2.22-11 and 2.22-12 provide a comparison between the fission product release fractions and the timing/duration of releases to the containment as assumed in TID-14844 and in RG 1.183.

- Instead of the iodine being primarily in the elemental form, the iodine is mainly in the form of cesium iodide, which exists as particulate. The iodine characterization from RG 1.183 is compared in Table 2.22-10 with that from Regulatory Guide 1.4.

For the containment leakage analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay or leakage from the containment. For the ECCS leakage analyses, all iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS.

Containment Modeling

The containment building is modeled as two discrete volumes: sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers. The containment volume is 2.344E6 ft^3 with a sprayed fraction of 85.9% of the total (2.014E6 ft^3).

The containment is assumed to leak at the design leak rate of 0.1% per day for the first 24 hours of the accident and then to leak at half that rate (0.05% per day) for the remainder of the 30 day period following the accident considered in the analysis.

Removal of Activity from the Containment Atmosphere

The removal of elemental iodine from the containment atmosphere is accomplished only by containment sprays and radioactive decay. The removal of particulates from the containment atmosphere is accomplished by containment sprays, sedimentation and radioactive decay. The noble gases and the organic iodine are subject to removal only by radioactive decay.

One train of the containment spray system is assumed to operate following the LOCA. Injection spray is credited starting at 120 seconds in the event. This is conservative since it results in earlier spray termination and there is little activity in the containment at the time the sprays start. When the RWST drains to a predetermined setpoint level, the system automatically switches to recirculation of sump liquid to provide a source for the sprays. The analysis assumed that the sprays are terminated 4.0 hours from the start of the event.

Containment Spray Removal of Elemental Iodine

The Standard Review Plan (Reference 7) identifies a methodology for the determination of spray removal of elemental iodine independent of the use of spray additive. The removal rate constant is determined by:

$$\lambda_s = 6K_gTF / VD$$

Where: K_g = Gas phase mass transfer coefficient, ft/min
T = Time of fall of the spray drops, min
F = Volume flow rate of sprays, ft^3/hr
V = Containment sprayed volume, ft^3
D = Mass-mean diameter of the spray drops, ft

The upper limit of 20 hr^{-1} specified for this model is applied in the analysis.

Removal of elemental iodine from the containment atmosphere is assumed to be terminated when the airborne inventory drops to 0.5 percent of the total elemental iodine released to the containment (this is a DF of 200). With the RG 1.183 source term methodology this is interpreted as being 0.5 percent of the total inventory of elemental iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases. In the analysis, this occurs just before 2.0 hours.

Containment Spray Removal of Particulates

Particulate spray removal is determined using the model described in Reference 7. The first order spray removal rate constant for particulates may be written as follows:

$$\lambda_p = 3hFE / 2Vd$$

Where: h = Drop Fall Height, ft
F = Spray Flow Rate, ft^3/hr
V = Volume Sprayed, ft^3
E = Single Drop Collection Efficiency
d = Drop Diameter, ft

The E/d term depends upon the particle size distribution and spray drop size. From Reference 8 it is conservative to use 10 m^{-1} (3.05 ft^{-1}) for E/d until the point is reached when the inventory in the atmosphere is reduced to 2% of its original (DF of 50). With the RG 1.183 source term methodology this is interpreted as being 2% of the total inventory particulate iodine that is released to the containment atmosphere over the duration of gap and in-vessel release phases.

The parameters from Table 2.22-14 and the appropriate conversion factors were used to calculate the particulate spray removal coefficients. A value of 3.94 hr^{-1} was used in the analysis. When the airborne inventory drops to 2 percent of the total particulate iodine released to the containment (this is a DF of 50) this removal coefficient is reduced by a factor of 10. In the analysis this occurs at 2.5 hours.

Sedimentation Removal of Particulates

During spray operation, credit is taken for sedimentation removal of particulates in the unsprayed region. After sprays are terminated , credit for sedimentation is taken in both the sprayed and unsprayed regions.

Based on the Containment Systems Experiments (CSE) which examined the air cleanup experienced through natural transport processes, it was found that a large fraction of the aerosols were deposited on the floor rather than on the walls indicating that sedimentation was the dominant removal process for the test (Reference 8). The CSE tests determined that there was a significant sedimentation removal rate even with a relatively low aerosol concentration. From Reference 8, even at an air concentration of 10 $\mu g/m^3$, the sedimentation removal coefficient was above 0.3 hr^{-1}. With 2.0 percent of particulates remaining airborne at the end of credited spray removal, there would be more than 10,000 $\mu g/m^3$ and an even higher sedimentation rate would be expected. As noted above the DF of 50 occurs at 2.5 hr. For the analysis the sedimentation removal coefficient is conservatively assumed to be only 0.2 hr^{-1}. It is also conservatively assumed that sedimentation removal does not continue beyond a DF of 1000.

ECCS Leakage

When ECCS recirculation is established following the LOCA, leakage is assumed to occur from ECCS equipment outside containment. There are two pathways considered for the ECCS recirculation leakage. One is the leakage directly into the Auxiliary Building and the other is back-leakage into the refueling water storage tank (RWST). Recirculation is initiated when the RWST has drained to the pre-determined setpoint level (at about 20 minutes).

In accordance with Reg. Guide 1.183 (Position 5.1 of Appendix A), it is assumed that the iodine is instantaneously mixed in the primary containment sump water at the time of release from the core.

<u>Leakage to the Auxiliary Building</u>
The total ECCS recirculation leakage into the Auxiliary Building is 1 gpm and begins at 20 minutes. There is 2% partitioning of iodine in the leakage. Of this leakage, 0.967 gpm is inside the are served by of the Reactor Auxiliary Building Emergency Exhaust System (RABEES) which filters out much of the iodine released to the atmosphere. The remaining 0.033 gpm is released outside of RABEES without filtration.

<u>Leakage to the RWST</u>
ECCS back-leakage to the RWST is assumed at a rate of 1.5 gpm. The iodine in the sump solution is assumed to all be in nonvolatile iodide or iodate form. However, when the solution leaks into the RWST, the iodine will be in an acidic solution such that there is the possibility of conversion of iodine compounds to form elemental iodine. The amount of iodine that will convert to the elemental form is dependent both on the concentration of iodine in the solution and the pH of the solution. The initial boron concentration in the RWST is ~2500 ppm. The initial pH of the RWST solution is determined to be ~4.5. The RWST water pH and iodine concentration are determined as a function of time. Figure 3.1 of NUREG-5950 (Reference 12) is used to determine the

amount of iodine becoming elemental based on pH and iodine concentration. With an RWST pH of 4.5 and the low iodine concentration, the fraction of conversion to elemental iodine is 2%. After 24 hours, the RWST liquid pH will exceed 6.0 and the indicated conversion to elemental iodine is essentially zero; however, the fraction is conservatively assumed to be 1% for the remainder of the accident duration.

Elemental iodine is volatile and will partition between the liquid and the air in the RWST gas space. The partition coefficient for elemental iodine is determined to be 28.2 using a relationship to solution temperature from Reference 12. This is modeled by the transfer of a portion of the flow going to the RWST liquid and a portion going to the RWST gas space. The modeling of the air flow out of the RWST is based on diurnal heating and cooling cycle. This model ignores the effect of the large heat sink provided by the mass of water in the tank that would tend to moderate the effects of the heating and cooling from the sunlight and atmospheric temperature variations. The transfer from the RWST gas space to the environment is calculated to be 5.9 cfm based on displacement by the inleakage and air expansion from the heating/cooling cycle.

In the current HNP design record (and in the prior conventional source term SGR/uprate submittals), the RWST back leakage dose path was determined to be inconsequential, such that no atmospheric dispersion factor for this release point/receptor pair has been developed or reported to the NRC. For this AST analysis, a significantly higher assumed unfiltered air inleakage value greatly magnifies any control room dose impact this release path might have. Therefore, an approximate X/Q was determined from the existing containment to Control Room Emergency Air Intake (CREAI) X/Q, and the distances from the containment and RWST to the limiting (South) CREAI. Site drawings indicate that the RWST is approximately two-thirds of the distance from the containment to the South CREAI. Examination of various references dealing with determination of X/Q shows that the value varies as the inverse of the product of an x-y plume spread factor, and a z-direction plume spread factor. Both of these plume spread factors are, in turn, dependent on the distance from the source to the receptor. Using this relationship, the HNP RWST release X/Q was determined to be (3/2 * 3/2), or 2.25 times larger than the currently defined containment release point X/Q.

Control Room Isolation

In the event of a large break LOCA, the SI setpoint will be reached shortly after event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the post-accident recirculation mode of operation. It is assumed that the SI setpoint is reached immediately at the start of the event; only the 15 second delay time for switching from normal to emergency operating mode is modeled. An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation.

Acceptance Criteria

The offsite dose limit for a LOCA is 25 rem TEDE per RG 1.183. This is the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The large break LOCA doses are:

Exclusion Area Boundary	7.53 rem TEDE
Low Population Zone	4.33 rem TEDE
Control Room	4.99 rem TEDE

The acceptance criteria are met.

The control room dose reported models 300 cfm unfiltered inleakage into the control room. As required by Reg. Guide 1.183, Section 4.2.1 on Control Room Dose Calculation Methodology, the control room dose contributions of the release plume, the containment building post-accident radionuclide inventory, and the control room HVAC filter shine doses were conservatively evaluated. The small incremental dose contributions from these sources are included in the total Control Room TEDE dose reported above, which meets the specified acceptance criteria.

The exclusion area boundary dose reported is for the worst two hour period, determined to be from 0.4 hours to 2.4 hours.

The integrated activity released to the atmosphere is given in Table 2.22-15.

2.22.4.3.2 Steam Generator Tube Rupture Accident Doses

Please refer to NSSS Licensing Report Section 6.3.3 for the discussion of this radiological accident.

2.22.4.3.3 Locked Rotor Accident Doses

An instantaneous seizure of a reactor coolant pump rotor is assumed to occur which rapidly reduces flow through the affected reactor coolant loop. Fuel clad damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

Input Parameters and Assumptions

A summary of input parameters and assumptions is provided in Table 2.22-16.

The analysis of the locked rotor radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix G (Locked Rotor) and RG 1.183, Appendix H (Rod Ejection) for the fuel melt model.

It is assumed that 8% of the fuel rods in the core suffer damage as a result of the locked rotor sufficient that all of their gap activity is released to the reactor coolant system. Additionally, 1% of the fuel rods are conservatively assumed to experience centerline melt. Eight percent of the total I-131 core activity is in the fuel-cladding gap. Ten percent of the total Kr-85 core activity is in the fuel-cladding gap. Five percent of other iodine isotopes and other noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. In the calculation of activity releases from the failed/melted fuel the maximum radial peaking factor of 1.73 was applied. All noble gas and alkali metal activity in the damaged fuel (both gap activity and activity contained in the melted fuel) is released to the primary coolant. All of the iodine contained in the gap of failed fuel and 50 percent of the iodine activity contained in the melted fuel are released to the reactor coolant system.

The iodine activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the locked rotor occurs is assumed to be 10% of the primary side concentration.

The amount of primary to secondary SG tube leakage is assumed to be equal to the Technical Specification limit of 1 gpm total.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

For HNP it was assumed that plant cooldown to RHR operating conditions can be accomplished within 8 hours after initiation of the locked rotor event. At 8 hours after the accident, the RHR System is assumed to be placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system.
A primary and secondary side mass and energy balance was used to calculate the steam released from the steam generators from 0 to 2 hours and from 2 to 8 hours.

Control Room Isolation

The control room HVAC is switched to the emergency post-accident recirculation mode after receiving a high radiation signal. The high radiation signal is reached at 3 seconds into the event. The control room HVAC is switched over to the emergency post-recirculation mode at 18 seconds (3 second signal initiation plus 15 second delay time for switching between modes). An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation. The 15-second delay to switch between modes was also assumed with the operator action. Thus 2 hours and 33 seconds was actually modeled for the time of operator action switchover to the pressurization mode.

Acceptance Criteria

The offsite dose limit for a locked rotor is 2.5 rem TEDE per RG 1.183. This is 10% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The locked rotor doses are:

Exclusion Area Boundary	1.89 rem TEDE
Low Population Zone	1.40 rem TEDE
Control Room	3.17 rem TEDE

The acceptance criteria are met.

The exclusion area boundary doses reported are for the worst two hour period, determined to be from 6.0 to 8.0 hours.

The reported control room dose is based on 500 cfm unfiltered inleakage into the control room.

The integrated activity released to the atmosphere is given in Table 2.22-17.

2.22.4.3.4 Single RCCA Withdrawal Accident Doses

A single RCCA rod is withdrawn from the reactor core due to a malfunction in the rod control system. The single RCCA withdrawal causes an insertion of positive reactivity that results in a power excursion transient. Fuel clad damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

Input Parameters and Assumptions

A summary of input parameters and assumptions is provided in Table 2.22-18.

The analysis of the single RCCA withdrawal radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix G (Locked Rotor) for secondary system leakage release path modeling and RG 1.183, Appendix H (Rod Ejection) for gap fraction and fuel melt models.

It is assumed that 4% of the fuel rods in the core suffer damage as a result of the event sufficient that all of their gap activity is released to the reactor coolant system. Additionally 1% of the fuel rods are conservatively assumed to experience centerline melt. Ten percent of the total core activity of iodine, ten percent of the total core activity for noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap and are released into the primary coolant. In the calculation of activity releases from the failed/melted fuel the maximum radial peaking factor of 1.73 was applied.

All noble gas and alkali metal activity in the damaged fuel (both gap activity and activity contained in the melted fuel) is released to the primary coolant. All of the iodine contained in the gap of failed fuel and 50 percent of the iodine activity contained in the melted fuel are released to the reactor coolant system.

The iodine activity concentration of the secondary coolant at the time the event occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the event occurs is assumed to be 10% of the primary side concentration.

The amount of primary to secondary SG tube leakage is assumed to be equal to the Technical Specification limit of 1 gpm total.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

For HNP it was assumed that plant cooldown to RHR operating conditions can be accomplished within 8 hours after initiation of the locked rotor event. At 8 hours after the accident, the RHR System is assumed to be placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system.
A primary and secondary side mass and energy balance was used to calculate the steam released from the steam generators from 0 to 2 hours and from 2 to 8 hours.

Control Room Isolation

The control room HVAC is switched to the emergency post-accident recirculation mode after receiving a high radiation signal. The high radiation signal is reached at 3 seconds into the event. The control room HVAC is switched over to the emergency post-recirculation mode at 18 seconds (3 second signal initiation plus 15 second delay time for switching between modes). An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation. The 15-second delay to switch between modes was also assumed with the operator action. Thus 2 hours and 33 seconds was actually modeled for the time of operator action switchover to the pressurization mode.

Acceptance Criteria

The offsite dose limit for a single RCCA withdrawal accident is not defined in RG 1.183, however, the locked rotor offsite dose limit is 2.5 rem TEDE per RG 1.183. This is 10% of the guideline value of 10CFR50.67. Since the locked rotor event involves similar release mechanisms, its acceptance criteria will be assumed to apply to this single RCCA withdrawal accident. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The single RCCA withdrawal doses are:

Exclusion Area Boundary	1.57 rem TEDE
Low Population Zone	1.23 rem TEDE
Control Room	2.63 rem TEDE

The acceptance criteria are met.

The exclusion area boundary doses reported are for the worst two hour period, determined to be from 6.0 to 8.0 hours.

The control room dose reported models 500 cfm unfiltered inleakage into the control room.

The integrated activity released to the atmosphere is given in Table 2.22-19.

2.22.4.3.5 Main Steam Line Break Doses

The complete severance of a main steam line outside containment is assumed to occur. The affected SG will rapidly depressurize and release radioiodines initially contained in the secondary coolant and primary coolant activity, transferred via SG tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact SG and noble gas activity due to tube leakage is released to atmosphere through either the atmospheric dump valves (ADV) or the safety valves (MSSVs). The steam line break outside containment will bound any break inside containment since the outside break provides a means for direct release to the environment. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from this release.

Input Parameters and Assumptions

The analysis of the steam line break (SLB) radiological consequences uses the analytical methods and assumptions outlined in the RG 1.183, Appendix E. A summary of input parameters and assumptions is provided in Table 2.22-20.

For the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the SLB and has raised the RCS iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of dose equivalent (DE) I-131. For the accident-initiated iodine spike case, the reactor trip associated with the SLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 12 percent of the iodine in the fuel-clad gap, the gap inventory would be depleted within 5.0 hours and the spike is terminated at that time.

In addition to the iodine spiking cases, a fuel damage case was also considered. For the fuel failure case 1% of the fuel is assumed to fail releasing its gap activity. Iodines, noble gases and alkali metals are considered in the fuel failure case. All of the gap activity is released from the fuel. Additionally, the radial peaking factor of 1.73 was applied to the failed fuel inventory as indicated in RG 1.183.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a one-percent fuel defect level. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity. The iodine activity concentration of the secondary coolant at the time the SLB occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131.

The amount of primary to secondary SG tube leakage is assumed to be equal to the Technical Specification limit for 1 gpm total. The primary to secondary SG tube leakage is apportioned between the affected and unaffected SGs to result in the most conservative result. Leakage to the affected (ruptured) SG is directly released to the atmosphere thus using 0.35 gpm to the affected SG and 0.65 gpm to the two unaffected SGs would maximize the dose.

The SG connected to the broken steam line is assumed to boil dry within the initial two minutes following the SLB. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine initially in this SG is released to the environment. In addition, iodine carried over to the faulted SG by tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the SG.

In the intact SGs an iodine partition factor of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Eight hours after the accident, the RHR System is assumed to be placed into service for heat removal. After eight hours there are no further steam releases to the atmosphere from the intact steam generators.

Within 40 hours after the accident, the reactor coolant system has been cooled to below 212°F, and there are no further steam releases to atmosphere from the faulted steam generator.

Control Room Isolation

In the event of a SLB, the steamline pressure SI setpoint will be reached shortly after event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the post-accident recirculation mode of operation. It is assumed that SI signal is reached at 10 seconds. The control room HVAC switches from normal operation to post-accident recirculation mode of operation at 25 seconds (10 seconds for SI signal plus 15 second delay time). Two hours after the control room HVAC is in post-accident recirculation mode an operator action switches the control room to the pressurization mode.

Acceptance Criteria

The offsite dose limit for a SLB with a pre-accident iodine spike or fuel damage is 25 rem TEDE per RG 1.183. This is the guideline value of 10CFR50.67. For a SLB with an accident-initiated iodine spike, the offsite dose limit is 2.5 rem TEDE per RG 1.183. This is 10% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The SLB accident doses are listed below.

For the pre-accident iodine spike:

Exclusion Area Boundary	0.13 rem TEDE
Low Population Zone	0.14 rem TEDE
Control Room	0.36 rem TEDE

500 cfm unfiltered inleakage modeled into the control room.

For the accident-initiated iodine spike:

Exclusion Area Boundary	0.70 rem TEDE
Low Population Zone	1.04 rem TEDE
Control Room	2.47 rem TEDE

500 cfm unfiltered inleakage modeled into the control room.

For the fuel failure:

Exclusion Area Boundary	1.44 rem TEDE
Low Population Zone	2.52 rem TEDE
Control Room	3.95 rem TEDE

300 cfm unfiltered inleakage modeled into the control room.

The acceptance criteria are met.

The exclusion area boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours for the pre-accident iodine spike and for the fuel failure and from 5.0 to 7.0 hours for the accident-initiated iodine spike.

The integrated activity released to the atmosphere for the accident initiated iodine spike case, the pre-accident iodine spike case and the failed fuel case is given in Tables 2.22-21, 2.22-22, and 2.22-23 respectively.

2.22.4.3.6 Loss of Offsite Power Accident Doses

A loss of non-emergency AC power to plant auxiliaries would result in a turbine and reactor trip on loss of condenser vacuum. Heat removal from the secondary system would occur through the steam generator power-operated relief valves or safety valves. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or safety valves. In addition, iodine activity is contained in the secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

Input Parameters and Assumptions

A summary of input parameters and assumptions is provided in Table 2.22-24.

The analysis of the loss of offsite power (LOOP) radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix G (Locked Rotor) for secondary system leakage release path modeling and RG 1.183 Appendix E (Main Steam Line Break) for iodine spiking.

For the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the LOOP and has raised the RCS iodine concentration to the Technical Specification limit for a transient of 60 μCi/gm of dose equivalent (DE) I-131. For the accident-initiated iodine spike case, the reactor trip associated with the LOOP creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 12 percent of the iodine in the fuel-clad gap, the gap inventory would be conservatively depleted within 5.0 hours and the spike is terminated at that time.

The noble gas and alkali metal activity concentrations in the RCS at the time the accident occurs are based on a one percent fuel defect level. The iodine activity concentration of the secondary coolant at the time the LOOP occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the LOOP occurs is assumed to be 10% of the primary side concentration.

The amount of primary to secondary SG tube leakage is assumed to be equal to the Technical Specification limit of 1 gpm total.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

For HNP it was assumed that plant cooldown to RHR operating conditions can be accomplished within 8 hours after initiation of the LOOP event. At 8 hours after the accident, the RHR System is assumed to be placed into service for heat removal and there is no further steam release to the atmosphere from the secondary system.
A primary and secondary side mass and energy balance was used to calculate the steam released from the steam generators from 0 to 2 hours and from 2 to 8 hours.

Control Room Isolation

The control room HVAC is switched to the emergency post-accident recirculation mode after receiving a high radiation signal. The high radiation signal is reached at 3 seconds into the event. The control room HVAC is switched over to the emergency post-recirculation mode at 18 seconds (3 second signal initiation plus 15 second delay time for switching between modes). An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation. The 15-second delay to switch between modes was also assumed with the operator action. Thus 2 hours and 33 seconds was actually modeled for the time of operator action switchover to the pressurization mode.

<u>Acceptance Criteria</u>

The offsite dose limit for the loss of offsite power accident is not defined in RG 1.183. The offsite dose limit for a SLB with a pre-accident iodine spike or fuel damage is 25 rem TEDE per RG 1.183. This is the guideline value of 10CFR50.67. For a SLB with an accident-initiated iodine spike, the offsite dose limit is 2.5 rem TEDE per RG 1.183. This is 10% of the guideline value of 10CFR50.67. Since the SLB event involves similar iodine spiking, its acceptance criteria will be assumed to apply to this LOOP accident. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

<u>Results and Conclusions</u>

The LOOP accident doses are listed below.

For the pre-accident iodine spike:

Exclusion Area Boundary	0.012 rem TEDE
Low Population Zone	0.0092 rem TEDE
Control Room	0.028 rem TEDE

500 cfm unfiltered inleakage modeled into the control room.

For the accident-initiated iodine spike:

Exclusion Area Boundary	0.043 rem TEDE
Low Population Zone	0.022 rem TEDE
Control Room	0.065 rem TEDE

500 cfm unfiltered inleakage modeled into the control room.

The acceptance criteria are met.

The exclusion area boundary doses reported are for the worst two hour period, determined to be from 6.0 to 8.0 hours.

The integrated activity released to the atmosphere is given in Table 2.22-25 for the accident initiated iodine spike case and in Table 2.22-26 for the pre-accident iodine spike case.

2.22.4.3.7 Rod Ejection Accident Doses

It is assumed that a mechanical failure of a control rod mechanism pressure housing has occurred, resulting in the ejection of a rod cluster control assembly and drive shaft. As a result of the accident, some fuel clad damage and a small amount of fuel melt are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves or the main steam safety valves. Iodine and alkali metals group activity is contained in the secondary coolant prior to the accident, and some of this activity is released to the atmosphere as a result of steaming the steam generators following the accident. Finally, radioactive reactor coolant is discharged to the containment via the spill from the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

Input Parameters and Assumptions

The analysis of the rod ejection radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix H. Separate calculations are performed to calculate the dose resulting from the release of activity to containment and subsequent leakage to the environment and the dose resulting from the leakage of activity to the secondary system and subsequent release to the environment. The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths and nuclides considered. A summary of input parameters and assumptions is provided in Table 2.22-27.

Source Term

In determining the offsite doses following a rod ejection accident, it is assumed that 4% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released and that 2% of the fuel in the core melts. Ten percent of the total core activity of iodine and noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. In the calculation of activity releases from the failed/melted fuel the maximum radial peaking factor of 1.73 was applied.

For both the containment leakage release path and the primary to secondary leakage release path all noble gas and alkali metal activity contained in the failed fuel gap and in the melted fuel is available for release.

For the containment leakage release path all of the iodine contained in the failed fuel gap and 25 percent of the activity contained in the melted fuel is available for release.

For the primary to secondary leakage release path all of the iodine contained in the failed fuel gap and 50 percent of the activity contained in the melted fuel is available for release from the reactor coolant system.

Prior to the accident the iodine activity concentration of the primary coolant is 1.0 μCi/gm of DE I-131. The noble gas and alkali metal activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level. The iodine activity concentration of the secondary coolant at the time the rod ejection occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the rod ejection occurs is assumed to be 10% of the primary side concentration.

Iodine in containment is assumed to be 4.85% elemental, 0.15% organic and 95% particulate.

Iodine released from the secondary system is assumed to be 97% elemental and 3% organic.

Containment Release Pathway

The containment is assumed to leak at the design leak rate of 0.1% per day for the first 24 hours of the accident and then to leak at half that rate (0.05% per day) for the remainder of the 30 day period following the accident considered in the analysis.

For the containment leakage pathway, no credit is taken for plateout onto containment surfaces or for containment spray operation which would remove airborne particulates and elemental iodine. Sedimentation of alkali metal particulates in containment is credited.

Primary to Secondary Leakage Release Pathway

When determining doses due to the primary to secondary steam generator tube leakage, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment). The primary to secondary tube leakage and steaming from the steam generators continues until the reactor coolant system pressure drops below the secondary pressure. A conservative time of 2 hours was used for this analysis, although analyses of the small break LOCA pressure transient have shown that this would occur well before that time. The rod ejection pressure transient is similar to that of a small break LOCA.

The amount of primary to secondary SG tube leakage is assumed to be equal to the Technical Specification limit of 1 gpm total. Although the primary to secondary pressure differential drops throughout the event, the constant flow rate is conservatively maintained.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Control Room Isolation

In the rod ejection accident, the SI setpoint will be reached within 30 seconds from event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the post-accident recirculation mode of operation. A 15-second delay for the control room to switch between normal and post-accident recirculation modes is modeled. An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation.

Acceptance Criteria

The offsite dose limit for a rod ejection is 6.3 rem TEDE per RG 1.183. This is ~25% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The rod ejection doses are:

Exclusion Area Boundary	3.90 rem TEDE
Low Population Zone	4.00 rem TEDE
Control Room	4.30 rem TEDE

300 cfm unfiltered inleakage modeled into the control room.

The acceptance criteria are met.

The exclusion area bounday doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

The integrated activity released to the atmosphere is given in Table 2.22-28.

2.22.4.3.8 Small Break Loss of Coolant Accident Doses

An abrupt failure of the primary coolant system is assumed to occur and it is assumed that the break is small enough that the containment spray system is not immediately actuated by high containment pressure but that the core experiences substantial cladding damage such that the fission product gap activity of all fuel rods is released. Activity that is released to the containment is assumed to be released to the environment due to the containment leaking at its design rate. There is also a release path through the steam generators (primary to secondary) until the primary system becomes depressurized to below the secondary system pressure.

Input Parameters and Assumptions

The analysis of the SBLOCA radiological consequences uses the analytical methods and assumptions outlined in RG 1.183, Appendix H (Rod Ejection) for release path modeling and for fuel melt release fractions for iodines and RG 1.183, Table 3 for gap fractions.

Separate calculations are performed to calculate the dose resulting from the release of activity to containment and subsequent leakage to the environment and the dose resulting from the leakage of activity to the secondary system and subsequent release to the environment. The total offsite and control room doses are the sum of the doses resulting from each of the postulated release paths and nuclides considered. A summary of input parameters and assumptions is provided in Table 2.22-29.

Source Term

In determining the offsite doses following a SBLOCA, it is assumed that all of the fuel rods in the core suffer sufficient damage that all of their gap activity is released and that 2% of the fuel in the core melts. Eight percent of the total I-131 core activity is in the fuel-cladding gap. Ten percent of the total Kr-85 core activity is in the fuel-cladding gap. Five percent of other iodine isotopes and other noble gases and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap.

Position 3.1 of RG 1.183 indicates that for accidents involving the entire core the radial peaking factor should be applied. Since 100% of the rods are assumed to be damaged, this guidance does not apply to the gap release but does apply to the small fraction of the core involved in fuel melt. In the calculation of activity releases from the 2% melted fuel the maximum radial peaking factor of 1.73 was applied.

For both the containment leakage release path and the primary to secondary leakage release path all noble gas and alkali metal activity in the failed fuel gap and in the melted fuel is available for release.

For the containment leakage release path all of the iodine in the failed fuel gap and 25 percent of the activity in the melted fuel is available for release from containment.

For the primary to secondary leakage release path all of the iodine failed fuel gap and 50 percent of the activity in the melted fuel is available for release from the reactor coolant system.

Prior to the accident the iodine activity concentration of the primary coolant is 1.0 μCi/gm of DE I-131. The noble gas and alkali metal activity concentration in the RCS at the time the accident occurs is based on a one percent fuel defect level. The iodine activity concentration of the secondary coolant at the time the SBLOCA occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131. The alkali metal activity concentration of the secondary coolant at the time the SBLOCA occurs is assumed to be 10% of the primary side concentration.

Iodine in containment is assumed to be 4.85% elemental, 0.15% organic and 95% particulate.

Iodine released from the secondary system is assumed to be 97% elemental and 3% organic.

Containment Modeling

The containment building is modeled as two discrete volumes: sprayed and unsprayed. The volumes are conservatively assumed to be mixed only by the containment fan coolers. The containment volume is 2.344E6 ft^3 with a sprayed fraction of 85.9% of the total (2.014E6 ft^3).

The containment is assumed to leak at the design leak rate of 0.1% per day for the first 24 hours of the accident and then to leak at half that rate (0.05% per day) for the remainder of the 30 day period following the accident considered in the analysis.

Removal of Activity from the Containment Atmosphere

The removal of elemental iodine from the containment atmosphere is accomplished only by containment sprays and radioactive decay. The removal of particulates from the containment atmosphere is accomplished by containment sprays, sedimentation and radioactive decay. The removal of elemental iodine and particulates with containment sprays is modeled in the sprayed region only. The noble gases and the organic iodine are subject to removal only by radioactive decay.

One train of the containment spray system is assumed to operate following the SBLOCA. Injection spray is credited starting at 30 minutes into the event. The analysis assumed that the sprays are terminated after 30 minutes of operation. The analysis used an elemental iodine spray removal coefficient of 20 hr^{-1} until the DF limit of 200 is reached after 15 minutes of spray operation. The particulate iodine spray removal coefficient of 3.94 hr^{-1} is credited for the entire 30 minutes of spray. These spray removal coefficients are the same as those used in the large break LOCA analysis discussed in Section 2.22.4.3.1.

After spray termination, credit is taken for sedimentation removal of particulates in both the sprayed and unsprayed regions of containment. The sedimentation removal coefficient of 0.2/hr is the same as that used in the large break LOCA analysis discussed in Section 2.22.4.3.1.

Primary to Secondary Leakage Release Pathway

When determining doses due to the primary to secondary steam generator tube leakage, all the iodine, alkali metals group and noble gas activity (from prior to the accident and resulting from the accident) is assumed to be in the primary coolant (and not in the containment). The primary to secondary tube leakage and the steaming from the steam generators continue until the reactor coolant system pressure drops below the secondary pressure. A conservative time of 2 hours was used for this analysis, although analyses of the small break LOCA pressure transient have shown that this would occur well before that time.

The amount of primary to secondary SG tube leakage is assumed to be equal to the Technical Specification limit of 1 gpm total. Although the primary to secondary pressure differential gradually drops, the constant flow rate is conservatively maintained until primary pressure is lower than secondary pressure.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is also applied to alkali metals.

All noble gas activity carried over to the secondary side through SG tube leakage is assumed to be immediately released to the outside atmosphere.

Control Room Isolation

The SI setpoint will be reached within 60 seconds from event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the post-accident recirculation mode of operation. A 15-second delay for the control room to switch between normal and post-accident recirculation modes is modeled. An operator action switches the control room from the post-accident recirculation mode to the pressurization mode at 2 hours after event initiation.

Acceptance Criteria

The offsite dose limit for a LOCA is 25 rem TEDE per RG 1.183. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The SBLOCA doses are:

Exclusion Area Boundary	9.24 rem TEDE
Low Population Zone	2.83 rem TEDE
Control Room	4.10 rem TEDE

300 cfm unfiltered inleakage modeled into the control room.

The acceptance criteria are met.

The exclusion area boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

The integrated activity released to the atmosphere is given in Table 2.22-30.

2.22.4.3.9 Fuel Handling Accident in Containment Doses

A fuel assembly is assumed to be dropped in containment and damaged during refueling. Activity released from the damaged assembly is released to the outside atmosphere through the containment openings (such as the personnel air lock door or the equipment hatch).

Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 2.22-31. This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours. The pool referred to in RG 1.183 is interpreted as the flooded reactor cavity for the purposes of evaluating the fuel handling accident in containment. No credit is taken for isolation of containment for the FHA in containment.

Source Term

Consistent with Regulatory Guide 1.183 (Position 1.2 of Appendix B), the radionuclides considered are xenons, kryptons, halogens, cesiums and rubidiums. The list of xenons, kryptons, and halogens considered is given in Table 2.22-31. The cesium and rubidium are not included because they are not assumed to be released from the pool as discussed later.

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, 10% for Kr-85, and 5% for all other nuclides.

As in the existing licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly (264 rods) are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.73 times the core average power.

The decay time used in the analysis is 100 hours. Thus, the analysis supports the design basis limit of 100 hours decay time prior to fuel movement.

Fission Product Form

In accordance with RG 1.183 iodine species in the pool is 99.85% elemental and 0.15% organic. This is based on the split leaving the fuel of 95% cesium iodide (CsI), 4.85% elemental iodine and 0.15% organic iodine. It assumed that all CsI is dissociated in the water and re-evolves as elemental. This is assumed to occur instantaneously. Thus, 99.85% of the iodine released is elemental.

Pool Scrubbing Removal of Activity

Reg. Guide 1.183 (Reference 2) provides that for 23 feet of water above the fuel, or greater, the DF for elemental and organic iodine are 500 and 1, respectively. The Reg. Guide goes on to say that this results in an overall effective DF of 200. Thus, in accordance with the guidance cited in RG, the numerical result for overall effective DF is approximately 286. The overall effective DF of 200, therefore, represents a conservative approximation of the results of the detailed calculation. It was determined that for the HNP specific water height above the failed fuel in the containment of 22 feet, the elemental DF would be at least 382, instead of the Reg. Guide allowable elemental DF of 500. Using the elemental DF of 382, it was determined that the overall effective DF for 22 feet of coverage would be 243. Since this continues to exceed the Reg. Guide cited overall effective DF of 200, it remains conservative to use the overall DF of 200 in the HNP dose calculations.

Using an overall DF of 200 gives an elemental DF of 286. The iodine chemical split above the pool is 70% elemental and 30% organic. This is different than the RG 1.183 stated value for the iodine chemical split above the pool of 57% elemental and 43% organic which is based on an elemental DF of 500.

However, the split between elemental and organic iodine leaving the pool has no impact on the analysis since the control room filter efficiencies for the two iodine forms are the same, and no other filtration is credited.

The cesium and rubidium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

Isolation and Filtration of Release Paths

No credit is taken for removal of iodine by filters nor is credit taken for isolation of release paths.

Although the containment purge will be automatically isolated on a purge line high radiation alarm, isolation is not modeled in the analysis. The activity released from the damaged assembly is assumed to be released to the outside atmosphere over a 2 hour period. Since no filters or containment isolation is modeled, this analysis supports refueling operation with the equipment hatch or personnel air lock remaining open.

Control Room Isolation

It is assumed that the control room HVAC system is initially operating in normal mode. The activity level in the intake duct causes a high radiation signal almost immediately. It is conservatively assumed that the post accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room HVAC enters pressurization mode due to operator action at 2 hours after isolation signal.

Acceptance Criteria

The offsite dose limit for a fuel handling accident is 6.3 rem TEDE per RG 1.183. This is ~25% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The fuel handling accident in containment doses are:

Exclusion Area Boundary	2.03 rem TEDE
Low Population Zone	0.46 rem TEDE
Control Room	1.39 rem TEDE

500 cfm unfiltered inleakage modeled into the control room.

The acceptance criteria are met.

The amount of activity released to the atmosphere is given in Table 2.22-32.

2.22.4.3.10 Fuel Handling Accident In Fuel Building Doses

A fuel assembly is assumed to be dropped inside the fuel handling building and damaged during refueling. Activity released from the damaged assembly is released to the outside atmosphere through the fuel pool ventilation system.

<u>Input Parameters and Assumptions</u>

The major assumptions and parameters used in the analysis are itemized in Table 2.22-33. This analysis involves dropping a recently discharged (100 hour decay) PWR fuel assembly onto 52 Brunswick BWR fuel assemblies. This analysis also includes 50 PWR rods additionally damaged in the accident. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours.

Source Term

Consistent with Regulatory Guide 1.183 (Position 1.2 of Appendix B), the radionuclides considered are xenons, kryptons, halogens, cesiums and rubidiums. The list of xenons, kryptons, and halogens considered is given in Table 2.22-33. The cesium and rubidium are not included because they are not assumed to be released from the pool as discussed later.

The calculation of the radiological consequences following a FHA uses gap fractions of 8% for I-131, 10% for Kr-85, and 5% for all other nuclides.

As in the existing licensing basis, it is assumed that all of the fuel rods in the equivalent of one fuel assembly plus 50 additional PWR rods (314 rods) plus 52 Brunswick BWR fuel assemblies are damaged to the extent that all their gap activity is released. The assembly inventory is based on the assumption that the PWR fuel assembly has been operated at 1.73 times the core average power and the BWR fuel assemblies have been operated at 1.5 times the core average power.

The BWR fuel inventory was conservatively evaluated at the IF-300 shipping cask limits recently approved in Reference 11. The decay time used in the analysis is 100 hours for the PWR fuel and 4 years for the BWR fuel. Thus, the analysis supports the design basis limit of 100 hours decay time prior to fuel movement.

Fission Product Form

Iodine species in the pool is 99.85% elemental and 0.15% organic. This is based on the split leaving the fuel of 95% cesium iodide (CsI), 4.85% elemental iodine and 0.15% organic iodine. It assumed that all CsI is dissociated in the water and re-evolves as elemental. This is assumed to occur instantaneously. Thus, 99.85% of the iodine released is elemental.

Pool Scrubbing Removal of Activity

RG 1.183 (Reference 2) provides that for 23 feet of water above the fuel, or greater, the DF for elemental and organic iodine are 500 and 1, respectively. The Reg. Guide goes on to say that this results in an overall effective DF of 200. Thus, in accordance with the guidance cited in RG 1.183, the numerical result for overall effective DF is approximately 286. The overall effective DF of 200, therefore, represents a conservative approximation of the results of the detailed calculation. It was determined that for the HNP specific water height above the failed fuel in the fuel handling building of 21 feet, the elemental DF would be at least 291, instead of the Reg. Guide allowable elemental DF of 500. Using the elemental DF 291, it was determined that the overall effective DF for 21 feet of coverage would be 203. Since this continues to exceed the Reg. Guide cited overall effective DF of 200, it remains conservative to use the overall DF of 200 in the HNP dose calculations.

Using an overall DF of 200 gives an elemental DF of 286. The iodine chemical split above the pool is 70% elemental and 30% organic. This is different than the RG 1.183 stated value for the iodine chemical split above the pool of 57% elemental and 43% organic which is based on an elemental DF of 500.

However, the split between elemental and organic iodine leaving the pool has no impact on the analysis since the control room filter efficiencies for the two iodines are the same.

The cesium and rubidium released from the damaged fuel rods is assumed to remain in a nonvolatile form and would not be released from the pool.

Isolation and Filtration of Release Paths

Credit is taken for removal of iodine by filters by the spent fuel pool ventilation system operation. Credit is not taken for isolation of release paths.

The activity released from the damaged assembly is assumed to be released to the fuel building and subsequently to the atmosphere over a 2 hour period.

Control Room Isolation

It is assumed that the control room HVAC system begins in normal mode. The activity level in the intake duct causes a high radiation signal almost immediately. It is conservatively assumed that the post accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room HVAC is placed into pressurization mode at 2 hours after isolation signal.

Acceptance Criteria

The offsite dose limit for a fuel handling accident is 6.3 rem TEDE per RG 1.183. This is ~25% of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The fuel handling accident in fuel building doses are:

Exclusion Area Boundary	0.34 rem TEDE
Low Population Zone	0.077 rem TEDE
Control Room	0.12 rem TEDE

500 cfm unfiltered inleakage modeled into the control room.

The acceptance criteria are met.

The amount of activity released to the atmosphere is given in Table 2.22-34.

2.22.4.3.11 Letdown Line Break Accident Doses

The most severe radioactivity release from a failed line carrying primary coolant outside of containment is the rupture of the letdown line. For such a break, the reactor coolant letdown flow would have passed from the cold leg and through the regenerative heat exchanger. This failure causes a direct release pathway from the primary system to the environment until the break can be isolated.

Input Parameters and Assumptions

The analysis of the small line break outside of containment (SLBOC) radiological consequences uses the analytical methods and assumptions outlined in SRP 15.6.2 (Reference 13) since this accident is not discussed in RG 1.183. A summary of input parameters and assumptions is provided in Table 2.22-35.

The SRP indicates that accident-initiated iodine spiking be modeled. The accident-initiated iodine spike increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 12 percent of the iodine in the fuel-clad gap, the gap inventory would be depleted within 5.0 hours and the spike is terminated at that time.

The noble gas activity concentration in the RCS at the time the accident occurs is based on a one-percent fuel defect level. This is approximately equal to the Technical Specification value of 100/E bar μCi/gm for gross radioactivity.

The transfer of the primary coolant to the environment through the letdown line break is 200 gpm until 30 minutes. The iodine flashing factor for the released letdown flow is 0.4. Therefore of the iodine contained in the water released in the letdown line break, only 40% of the iodine is released to the auxiliary building atmosphere and of that all is released to the environment.

Control Room Isolation

It is assumed that the control room HVAC system begins in normal mode. The activity level in the intake duct causes a high radiation signal almost immediately. It is conservatively assumed that the post accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room is assumed to be placed in pressurization mode at 2 hours after isolation signal.

Acceptance Criteria

The offsite dose acceptance criteria from SRP 15.6.2 (Ref. 13) is designated as 10% of 10CFR100 limits. Applying this same basis to the 25 rem TEDE in 10CFR50.67, the limit for offsite doses is 2.5 rem TEDE. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The SLBOC doses are:

Exclusion Area Boundary	2.34 rem TEDE
Low Population Zone	0.53 rem TEDE
Control Room	1.50 rem TEDE

500 cfm unfiltered inleakage modeled into the control room.

The acceptance criteria are met.

The amount of activity released to the atmosphere is given in Table 2.22-36.

2.22.4.3.12 Waste Gas Decay Tank Rupture Accident Doses

For the gas decay tank rupture, a failure is assumed that results in the release of the contents of one gas decay tank.

Input Parameters and Assumptions

The major assumptions and parameters used to determine the doses due to the gas decay tank failure are given in Table 2.22-37.

The inventory of gases in the tank are provided in Table 2.22-37. A failure in the gaseous waste processing system is assumed to result in release of the tank inventory with a release duration of 2 hours. This failure causes a direct release pathway from the waste gas decay tank to the environment

Control Room Isolation

It is assumed that the control room HVAC system begins in normal operational mode. The activity level causes a high radiation signal almost immediately. It is conservatively assumed that the post accident recirculation control room HVAC mode is entered 15 seconds after event initiation. The control room is assumed to be placed in the pressurization mode by operator action at 2 hours after isolation signal.

Acceptance Criteria

The offsite dose limit for a gas decay tank rupture is defined in HNP Technical Specification 6.8.4j as 0.5 rem whole body. This translates to a dose limit of 0.5 rem TEDE. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

Results and Conclusions

The gas decay tank rupture doses are:

Exclusion Area Boundary	0.30 rem TEDE
Low Population Zone	0.069 rem TEDE
Control Room	0.049 rem TEDE

500 cfm unfiltered inleakage modeled into the control room.

The acceptance criteria are met.

The exclusion area boundary dose reported is for the worst two hour period, determined to be from 0.0 to 2.0 hours.

The amount of activity released to the atmosphere is given in Table 2.22-38.

2.22.5 Conclusions

The existing plant design, radiation protection measures, procedures and operating practices combine to keep onsite and general public exposures within regulatory limits and industry guidelines in accordance with the FSAR.

No changes or additions to structures, equipment, or procedures are necessary to provide adequate radiation protection for the operators or the public during normal or post-accident operations to support the SGR/Uprate. The existing structures, systems, and components can safely handle the changes in post accident source terms and releases from the SGR/Uprate conditions, and resulting onsite and offsite doses are less than the 10CFR guidelines and are within the SRP recommendations. Therefore, the radiological consequence acceptance criteria for postulated Condition II, III, and IV events are satisfied. These results are consistent with the current design and licensing bases discussed in the FSAR.

The results obtained with the Delta 75 RSGs at the uprated NSSS power level of 2912.4 MWt bound operation with the Delta 75 RSGs at the current NSSS thermal power level of 2787.4 MWt.

NUREG-1465 defines an alternate source term model for use in evaluating the radiological consequences of a postulated large break Loss-of-Coolant Accident with core melt. This alternative source term model also forms the basis for determining the radiological consequences for other design basis accidents as provided in Regulatory Guide 1.183.

Full implementation of this alternative source term methodology (as defined in Regulatory Guide 1.183) into the Shearon Harris Nuclear Power Plant's design basis accident analysis has been made to support potential changes in the plant design and operation. Analyses of the radiological consequences of the large break LOCA, steam generator tube rupture, locked rotor, single RCCA withdrawal, loss of offsite power, rod ejection, small break LOCA, steamline break, fuel handling accident, letdown line break outside containment, and gas decay tank rupture have been made using the Regulatory Guide 1.183 methodology. The analyses used assumptions consistent with proposed changes in plant design and operation and the calculated doses do not exceed the defined acceptance criteria.

This report supports the following changes to Shearon Harris Nuclear Power Plant's design and operation:

- Movement of fuel in the containment with the equipment hatch and/or personnel air lock open.
- Increase of the Technical Specification limit for primary coolant iodine activity from the October 2000 Steam Generator Replacement Submittal proposed value of 0.35 μCi/gm DE I-131 to 1.0 μCi/gm DE I-131. Note that 1.0 μCi/gm DE I-131 is the value that currently exists in the approved HNP Tech Specs for pre-SGR/PUR operation.
- Allowable unfiltered inleakage into the control room of 300 cfm.

2.22.6 References

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2. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.
3. NRC Final Rule 10CFR50.67, issued in Federal Register, Vol. 64, No. 246, pages 71990-72002, 12/23/99.
4. U.S. Nuclear Regulatory Commission NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.
5. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88—0202, September 1988.
6. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, 1979.
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8. Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983

9. Murphy, K. G., Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Proceedings of the Thirteenth AEC Air Cleaning Conference held August 1974, published March 1975.

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11. IF-300 Cask License, Certificate of Compliance #9001, New Appendix D, NEDO-10084-5.

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13. NUREG-0800, Standard Review Plan 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, July 1981.

14. International Commission on Radiological Protection, "Radionuclide Transformations, Energy and Intensity of Emissions," ICRP Publication 38, 1983.

15. HNP Final Safety Analysis Report

16. HNP Technical Specifications

3/4.4.8	Specific Activity
Table 3.3-6	Radiation Monitoring Instrumentation for Plant Operations
6.8.4j	Offsite Dose Limit for a Gas Decay Tank Rupture
6.11	Radiation Protection Program
6.12	High Radiation Area

17. NUREG-1038, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant (Units 1 and 2)," dated November 1983.

18. NUREG-1038, Supplement No. 1, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1984.

19. NUREG-1038, Supplement No. 2, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated June 1985.

20. NUREG-1038, Supplement No. 3, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated May 1986.

21. NUREG-1038, Supplement No. 4, "Safety Evaluation Report Related to the Operation of the Shearon Harris Nuclear Power Plant Unit No. 1," dated October 1986.

22. 10CFR20, "Standards For Protection Against Radiation."

23. 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants."

    Criterion 19, "Control Room"
    Criterion 60" "Control of Radioactive Releases to the Environment"
    Criterion 64, "Monitoring Radioactive Releases"

24. 10CFR50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

25. 10CFR100, "Reactor Site Criteria."

26. RG 8.8, " Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," (Rev. 3, 6/78).

27. RG 1.4, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident," (Rev.2, 6 /74).

28. RG 1.49, "Power Levels of Nuclear Power Plants," (Rev. 1, 12/73).

29. RG 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants," (Rev. 2, 3/78).

30. RG 1.78, "Assumptions for Evaluating the Habitability Of A Nuclear Power Plant Control Room During A Postulated Hazardous Chemical Release," (Rev. 0, 6/74).

31. RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," (Rev.1, 10/77).

32. RG 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," (Rev. 1, 7/77).

33. RG 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," (Rev. 0-R, 5/77).

34. RG 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," (Rev. 1, 4/77).

35. RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," (Rev. 1, 10/79)

36. Branch Technical Position ESTB 11-5, "Postulated Radioactive Releases due to a Waste Gas System Leak or Failure."

37. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

Table 2.22-1: Committed Effective Dose Equivalent Dose Conversion Factors

Isotope	DCF (rem/curie)	Isotope	DCF (rem/curie)
I-131	3.29E4	Cs-134	4.62E4
I-132	3.81E2	Cs-136	7.33E3
I-133	5.85E3	Cs-137	3.19E4
I-134	1.31E2	Rb-86	6.63E3
I-135	1.23E3		
		Ru-103	8.95E3
Kr-83m	N/A	Ru-105	4.55E2
Kr-85m	N/A	Ru-106	4.77E5
Kr-85	N/A	Rh-105	9.56E2
Kr-87	N/A	Mo-99	3.96E3
Kr-88	N/A	Tc-99m	3.3E1
Xe-131m	N/A		
Xe-133m	N/A	Y-90	8.44E3
Xe-133	N/A	Y-91	4.89E4
Xe-135m	N/A	Y-92	7.80E2
Xe-135	N/A	Y-93	2.15E3
Xe-138	N/A	Nb-95	5.81E3
		Zr-95	2.37E4
Te-127	3.18E2	Zr-97	4.33E3
Te-127m	2.15E4	La-140	4.85E3
Te-129m	2.39E4	La-141	5.81E2
Te-129	9.0E1	La-142	2.53E2
Te-131m	6.4E3	Nd-147	6.85E3
Te-132	9.44E3	Pr-143	1.09E4
Sb-127	6.04E3	Am-241	4.44E8
Sb-129	6.44E2	Cm-242	1.73E7
		Cm-244	2.48E8
Ce-141	8.96E3		
Ce-143	3.39E3	Sr-89	4.14E4
Ce-144	3.74E5	Sr-90	1.3E6
Pu-238	3.92E8	Sr-91	1.66E3
Pu-239	4.3E8	Sr-92	8.1E2
Pu-240	4.3E8	Ba-139	1.7E2
Pu-241	8.26E6	Ba-140	3.74E3
Np-239	2.51E3		

Table 2.22-2: Effective Dose Equivalent Dose Conversion Factors

Isotope	DCF (rem· m^3/Ci· sec)	Isotope	DCF (rem· m^3/Ci· sec)
I-131	6.734E-2	Cs-134	0.2801
I-132	0.4144	Cs-136	0.3922
I-133	0.1088	Cs-137	0.1066
I-134	0.4810	Rb-86	1.780E-2
I-135	0.2953		
Kr-83m	5.550E-6	Ru-103	8.325E-2
Kr-85m	2.768E-2	Ru-105	0.1410
Kr-85	4.403E-4	Ru-106	0.0
Kr-87	0.1524	Rh-105	1.376E-2
Kr-88	0.3774	Mo-99	2.694E-2
Xe-131m	1.439E-3	Tc-99m	2.179E-2
Xe-133m	5.069E-3		
Xe-133	5.772E-3	Y-90	7.030E-4
Xe-135m	7.548E-2	Y-91	9.620E-4
Xe-135	4.403E-2	Y-92	4.810E-2
Xe-138	0.2135	Y-93	1.776E-2
		Nb-95	0.1384
Te-127	8.954E-4	Zr-95	0.1332
Te-127m	5.439E-4	Zr-97	3.337E-2
Te-129m	5.735E-3	La-140	0.4329
Te-129	1.018E-2	La-141	8.843E-3
Te-131m	0.2594	La-142	0.5328
Te-132	3.811E-2	Nd-147	2.290E-2
Sb-127	0.1232	Pr-143	7.770E-5
Sb-129	0.2642	Am-241	3.027E-3
		Cm-242	2.105E-5
Ce-141	1.269E-2	Cm-244	1.817E-5
Ce-143	4.773E-2		
Ce-144	3.156E-3	Sr-89	2.860E-4
Pu-238	1.806E-5	Sr-90	2.786E-5
Pu-239	1.569E-5	Sr-91	0.1277
Pu-240	1.758E-5	Sr-92	0.2512
Pu-241	2.683E-7	Ba-139	8.029E-3
Np-239	2.845E-2	Ba-140	3.175E-2

Table 2.22-3: Offsite Breathing Rates and Atmospheric Dispersion Factors

	Offsite Breathing Rates (m^3/sec)
0 - 8 hours	3.5E-4
8 - 24 hours	1.8E-4
>24 hours	2.3E-4

	Offsite Atmospheric Dispersion Factors (sec/m^3)
Exclusion Area Boundary*	6.17E-4
Low Population Zone	
0 - 2 hours	1.4E-4
2 - 24 hours	1.0E-4
1 - 4 days	5.9E-5
> 4 days	2.4E-5

* This exclusion area boundary atmospheric dispersion factor is conservatively applied during all time intervals in the determination of the limiting two hour period.

Table 2.22-4: Control Room Parameters

Volume (ft^3)	71,000
Normal Ventilation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	0.0
Unfiltered Makeup Flow Rate	1050.0
Unfiltered Recirculation Flow Rate	(Not modeled - no impact on analyses)
Post Accident Recirculation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	4000.0
Unfiltered Inleakage	Maximum allowed by accident (300 – 500)
Unfiltered Recirculation Flow Rate	(Not modeled - no impact on analyses)
Pressurization Mode Flow Rates (cfm)	
Filtered Makeup Air Flow Rate	400.0
Filtered Recirculation Flow Rate	3600.0
Unfiltered Inleakage	Maximum allowed by accident (300 – 500)
Unfiltered Recirculation Flow Rate	(Not modeled - no impact on analyses)
Filter Efficiencies (%)	
Elemental	99
Organic	99
Particulate	99
CR Radiation Monitor Sensitivity (µCi/ml for Xe-133)	3.0E-6
CR Radiation Monitor Location	Emergency & normal air intakes
Delay to Initiate Switchover of Post Accident signal Recirculation HVAC mode after radiation	15 seconds
Operator Action Time to Switch to Pressurization Mode	2 hours

Table 2.22-4 cont'd: Control Room Parameters

Breathing Rate - Duration of the Event (m^3/sec)	3.5E-4
Atmospheric Dispersion Factors (sec/m^3)	
0 – 8 hours	4.08E-3
8 – 24 hours	1.16E-3
1 – 4 days	3.25E-4
4 – 30 days	1.23E-5
Atmospheric Dispersion Factors for RWST vent release following a Large Break LOCA (sec/m^3)	
0 – 8 hours	9.18E-3
8 – 24 hours	2.61E-3
1 – 4 days	7.31E-4
4 – 30 days	2.77E-5
Occupancy Factors*	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4

* These occupancy factors (from Reference 9) have been conservatively incorporated in the atmospheric dispersion factors. This is conservative since it does not allow the benefit of reduced occupancy for activity already present in the control room from earlier periods.

Table 2.22-5: Not Used

Table 2.22-6: Not Used

Table 2.22-7: Core Total Fission Product Activities

Based on 102% of 2900 MWt

Isotope	Activity (Ci)	Isotope	Activity (Ci)
I-131	8.02E+07	Cs-134	1.53E+07
I-132	1.16E+08	Cs-136	4.27E+06
I-133	1.64E+08	Cs-137	9.17E+06
I-134	1.80E+08	Rb-86	1.78E+05
I-135	1.53E+08		
		Ru-103	1.22E+08
Kr-85	8.62E+05	Ru-105	8.39E+07
Kr-85m	2.19E+07	Ru-106	4.15E+07
Kr-87	4.22E+07	Rh-105	7.66E+07
Kr-88	5.95E+07	Mo-99	1.47E+08
Xe-131m	8.96E+05	Tc-99m	1.29E+08
Xe-133	1.60E+08		
Xe-133m	5.12E+06	Y-90	7.14E+06
Xe-135	3.83E+07	Y-91	1.04E+08
Xe-135m	3.21E+07	Y-92	1.08E+08
Xe-138	1.37E+08	Y-93	1.24E+08
		Nb-95	1.38E+08
Te-127	8.45E+06	Zr-95	1.37E+08
Te-127m	1.10E+06	Zr-97	1.36E+08
Te-129	2.53E+07	La-140	1.50E+08
Te-129m	3.76E+06	La-141	1.34E+08
Te-131m	1.16E+07	La-142	1.30E+08
Te-132	1.14E+08	Nd-147	5.38E+07
Sb-127	8.55E+06	Pr-143	1.22E+08
Sb-129	2.57E+07	Am-241	1.06E+04
		Cm-242	3.44E+06
Ce-141	1.35E+08	Cm-244	3.21E+05
Ce-143	1.25E+08		
Ce-144	1.01E+08	Sr-89	8.10E+07
Pu-238	2.58E+05	Sr-90	6.82E+06
Pu-239	2.38E+04	Sr-91	9.97E+07
Pu-240	3.26E+04	Sr-92	1.07E+08
Pu-241	1.02E+07	Ba-139	1.47E+08
Np-239	1.57E+09	Ba-140	1.42E+08

Table 2.22-8: RCS Coolant Concentrations

Based on 1% Fuel Defects

Isotope	Activity (μCi/gm)
I-131	1.71
I-132	2.47
I-133	7.234
I-134	5.67E-01
I-135	1.84
Kr-85m	1.73
Kr-85	1.06E+01
Kr-87	1.10
Kr-88	3.21
Xe-131m	3.41
Xe-133m	4.86
Xe-133	2.76E+02
Xe-135m	4.36E-01
Xe-135	8.52
Xe-138	6.30E-01
Cs-134	1.55
Cs-136	3.21
Cs-137	1.61
Rb-86	1.97E-02

Iodine concentrations are converted to dose equivalent (DE) I-131 using the dose conversion factors in ICRP-30 (Reference 6) for direct thyroid doses.

Table 2.22-9: Nuclide Decay Constants

Isotope	Decay Constant (hr^{-1})	Isotope	Decay Constant (hr^{-1})
I-131	0.00359	Cs-134	3.84E-5
I-132	0.303	Cs-136	2.2E-3
I-133	0.0333	Cs-137	2.64E-6
I-134	0.791	Rb-86	1.55E-3
I-135	0.105		
Kr-83m	0.379	Ru-103	7.35E-4
Kr-85m	0.155	Ru-105	0.156
Kr-85	7.37E-6	Ru-106	7.84E-5
Kr-87	0.547	Rh-105	1.96E-2
Kr-88	0.248	Mo-99	1.05E-2
Xe-131m	0.00241	Tc-99m	0.115
Xe-133m	0.0130		
Xe-133	0.00546	Y-90	1.08E-2
Xe-135m	2.72	Y-91	4.94E-4
Xe-135	0.0756	Y-92	0.196
Xe-138	2.93	Y-93	0.0686
		Nb-95	8.22E-4
Te-127	7.41E-2	Zr-95	4.51E-4
Te-127m	2.65E-4	Zr-97	4.1E-2
Te-129m	8.6E-4	La-140	1.72E-2
Te-129	0.598	La-141	0.176
Te-131m	2.31E-2	La-142	0.45
Te-132	8.86E-3	Nd-147	2.63E-3
Sb-127	7.5E-3	Pr-143	2.13E-3
Sb-129	0.16	Am-241	1.83E-7
		Cm-242	1.77E-4
Ce-141	8.89E-4	Cm-244	4.37E-6
Ce-143	0.021		
Ce-144	1.02E-4	Sr-89	5.72E-4
Pu-238	9.02E-7	Sr-90	2.72E-6
Pu-239	3.29E-9	Sr-91	0.073
Pu-240	1.21E-8	Sr-92	0.256
Pu-241	5.5E-6	Ba-139	0.502
Np-239	0.0123	Ba-140	2.27E-3

Table 2.22-10: Iodine Chemical Species

Iodine Form	RG 1.4	RG 1.183
Elemental	91%	4.85%
Organic	4%	0.15%
Particulate	5%	95%

Table 2.22-11: Fission Product Release Timing

Release Phase	Duration (TID-14844)	Duration (RG 1.183)[1]
Coolant Activity	instantaneous release	10 to 30 seconds
Gap Activity	instantaneous release	0.5 hour
Early In-vessel	instantaneous release	1.3 hour
Ex-vessel	not defined [2]	2 hours [3]
Late In-vessel	not defined [2]	10 hours [3]

1. Releases are sequential with the exception of the ex-vessel and the late in-vessel phases which both being at the end of the early in-vessel release phase.
2. Ex-vessel and late in-vessel release not defined in TID-14844.
3. Per RG 1.183, ex-vessel and late in-vessel releases are not applicable to design basis analyses.

Table 2.22-12: Core Fission Product Release Fractions

	Gap Release (1)		Early In-Vessel	
	TID	RG	TID	RG
Noble gases	n/a (2)	0.05	1.0	0.95
Halogens	n/a (2)	0.05	0.5 (3)	0.35
Alkali Metals	n/a	0.05	0.01 (4)	0.25
Tellurium group	n/a	0	0.01 (4)	0.05
Barium, Strontium	n/a	0	0.01 (4)	0.02
Noble Metals (Ruthenium group)	n/a	0	0.01 (4)	0.0025
Cerium group	n/a	0	0.01 (4)	0.0005
Lanthanides	n/a	0	0.01 (4)	0.0002

(1) The TID-14844 methodology does not specifically address the gap release. The RG 1.183 methodology assumes that gap and early in-vessel (core melt) releases are sequential. The TID-14844 source term model assumes the instantaneous release of 50% of core iodine and 100% of noble gases, with no distinction made between gap activity release and early in-vessel release. The RG 1.183 source term assumes a release of gap activity (5% of core) followed by the in-vessel release as defined.

(2) Gap fraction is not defined by TID-14844.

(3) Per TID-14844, half of this is assumed to plate out instantaneously.

(4) Referred to in TID-14844 as "other fission products" but not typically included in dose analyses.

Table 2.22-13: RG 1.183 Nuclide Groups

Group	Title	Elements in Group
1	Noble Gases	Xe, Kr
2	Halogens	I, Br
3	Alkali Metals	Cs, Rb
4	Tellurium Group	Te, Sb, Se
5	Barium, Strontium	Ba, Sr
6	Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
7	Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
8	Cerium Group	Ce, Pu, Np

Table 2.22-14: Assumptions Used for Large Break LOCA Dose Analysis

<u>Source Term</u>	
Core Activity	See Table 2.22-7
Activity release fractions and timing	See Tables 2.22-11 & 2.22-12
Iodine chemical form in containment (%)	
Elemental	4.85
Organic	0.15
Particulate (cesium iodide)	95
<u>Containment</u>	
Containment net free volume (ft^3)	2.344E6
Containment sprayed volume (ft^3)	2.014E6
Fan cooler units	
Number in operation	2
Flow rate (per unit)	31,250
Containment leak rates (weight %/day)	
0 – 24 hours	0.10
> 24 hours	0.05
Spray Operation	
Time to initiate sprays	120.0 seconds
Time to terminate spray operation	4.0 hours
Spray flow rates (gpm)	1730
Spray fall height (ft)	125
Removal Coefficients (hr^{-1})	
Spray elemental iodine removal	20.0
Spray particulate removal	3.94
Sedimentation particulate removal (after spray termination in sprayed region and from start of event in unsprayed region)	0.2

Table 2.22-14 cont'd: Assumptions Used for Large Break LOCA Dose Analysis

Containment sump volume (gal)	3.595E5
Time to initiate ECCS recirculation (min)	20
ECCS leak rate to Auxiliary Building (gpm)	1
Inside RABEES (gpm)	0.967
Outside RABEES (gpm)	0.033
ECCS leak rate to RWST (gpm)	1.5
Airborne fraction for ECCS leakage to RWST (%)	
Time < 24 hours and pH < 6.0	2.0
After 24 hours and pH ≥ 6.0	1.0
Partition Coefficient for Elemental Iodine for ECCS leakage to RWST	28.2
RABEES filter efficiencies (%)	
Elemental	95
Organic	95
Particulate	95

Table 2.22-15: Large Break LOCA Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	3.579E+02	5.287E+02	7.527E+02	1.577E+03	5.726E+03
I-132	3.621E+02	4.397E+02	4.464E+02	4.465E+02	4.465E+02
I-133	7.059E+02	1.013E+03	1.304E+03	1.647E+03	1.690E+03
I-134	3.269E+02	3.540E+02	3.541E+02	3.541E+02	3.541E+02
I-135	6.037E+02	8.174E+02	9.150E+02	9.341E+02	9.341E+02
Cs-134	5.220E+01	6.722E+01	7.265E+01	7.968E+01	1.393E+02
Cs-136	1.453E+01	1.869E+01	2.016E+01	2.188E+01	2.934E+01
Cs-137	3.129E+01	4.029E+01	4.354E+01	4.775E+01	8.348E+01
Rb-86	6.062E-01	7.799E-01	8.418E-01	9.165E-01	1.306E+00
Kr-85m	6.533E+02	3.270E+03	4.832E+03	4.903E+03	4.903E+03
Kr-85	3.214E+01	2.476E+02	8.219E+02	2.113E+03	1.321E+04
Kr-87	7.284E+02	1.768E+03	1.809E+03	1.809E+03	1.809E+03
Kr-88	1.555E+03	6.276E+03	7.626E+03	7.639E+03	7.639E+03
Xe-131m	3.329E+01	2.546E+02	8.290E+02	1.992E+03	6.736E+03
Xe-133m	1.873E+02	1.387E+03	4.163E+03	7.808E+03	1.015E+04
Xe-133	5.918E+03	4.484E+04	1.426E+05	3.163E+05	6.638E+05
Xe-135m	4.450E+01	4.668E+01	4.668E+01	4.668E+01	4.668E+01
Xe-135	1.280E+03	7.900E+03	1.599E+04	1.770E+04	1.771E+04
Xe-138	1.563E+02	1.619E+02	1.619E+02	1.619E+02	1.619E+02
Sr-89	1.800E+01	2.370E+01	2.485E+01	2.629E+01	3.652E+01
Sr-90	1.517E+00	1.998E+00	2.095E+00	2.221E+00	3.302E+00
Sr-91	2.013E+01	2.557E+01	2.612E+01	2.618E+01	2.618E+01
Sr-92	1.700E+01	2.022E+01	2.030E+01	2.030E+01	2.030E+01
Ba-139	1.705E+01	1.920E+01	1.921E+01	1.921E+01	1.921E+01
Ba-140	3.149E+01	4.142E+01	4.338E+01	4.567E+01	5.536E+01
Ce-141	9.156E-01	1.257E+00	1.379E+00	1.532E+00	2.506E+00
Ce-143	8.253E-01	1.118E+00	1.205E+00	1.251E+00	1.264E+00

Table 2.22-15 cont'd: Large Break LOCA Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr	24 hr	96 hr	720 hr
Ce-144	6.858E-01	9.418E-01	1.034E+00	1.154E+00	2.146E+00
Pu-238	1.752E-03	2.406E-03	2.643E-03	2.950E-03	5.590E-03
Pu-239	1.616E-04	2.220E-04	2.438E-04	2.722E-04	5.158E-04
Pu-240	2.214E-04	3.041E-04	3.340E-04	3.728E-04	7.065E-04
Pu-241	6.926E-02	9.513E-02	1.045E-01	1.166E-01	2.208E-01
Np-239	1.049E+01	1.428E+01	1.551E+01	1.643E+01	1.708E+01
Y-90	1.909E-02	2.602E-02	2.829E-02	3.011E-02	3.165E-02
Y-91	2.819E-01	3.870E-01	4.249E-01	4.729E-01	8.220E-01
Y-92	2.259E-01	2.799E-01	2.837E-01	2.837E-01	2.837E-01
Y-93	3.069E-01	4.038E-01	4.226E-01	4.249E-01	4.249E-01
Nb-95	3.738E-01	5.132E-01	5.632E-01	6.257E-01	1.034E+00
Zr-95	3.713E-01	5.098E-01	5.598E-01	6.232E-01	1.091E+00
Zr-97	3.492E-01	4.670E-01	4.961E-01	5.039E-01	5.043E-01
La-140	3.975E-01	5.398E-01	5.835E-01	6.105E-01	6.216E-01
La-141	2.877E-01	3.594E-01	3.653E-01	3.654E-01	3.654E-01
La-142	1.957E-01	2.251E-01	2.255E-01	2.255E-01	2.255E-01
Nd-147	1.454E-01	1.993E-01	2.184E-01	2.402E-01	3.243E-01
Pr-143	3.299E-01	4.525E-01	4.960E-01	5.470E-01	7.723E-01
Am-241	2.875E-05	3.948E-05	4.337E-05	4.841E-05	9.174E-05
Cm-242	9.327E-03	1.281E-02	1.407E-02	1.569E-02	2.878E-02
Cm-244	8.706E-04	1.196E-03	1.313E-03	1.466E-03	2.776E-03
Ru-103	3.393E+00	4.581E+00	5.008E+00	5.543E+00	9.137E+00
Ru-105	1.900E+00	2.364E+00	2.408E+00	2.409E+00	2.409E+00
Ru-106	1.155E+00	1.560E+00	1.707E+00	1.896E+00	3.485E+00
Rh-105	2.077E+00	2.771E+00	2.980E+00	3.097E+00	3.135E+00
Mo-99	4.035E+00	5.414E+00	5.865E+00	6.232E+00	6.556E+00
Tc-99m	3.084E+00	3.906E+00	4.015E+00	4.019E+00	4.019E+00

Table 2.22-15 cont'd: Large Break LOCA Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr	24 hr	96 hr	720 hr
Te-127m	6.115E-01	8.258E-01	9.032E-01	1.002E+00	1.780E+00
Te-127	4.259E+00	5.506E+00	5.737E+00	5.761E+00	5.761E+00
Te-129m	2.088E+00	2.819E+00	3.082E+00	3.409E+00	5.518E+00
Te-129	6.503E+00	7.228E+00	7.231E+00	7.231E+00	7.231E+00
Te-131m	6.255E+00	8.326E+00	8.928E+00	9.226E+00	9.295E+00
Te-132	6.265E+01	8.414E+01	9.129E+01	9.752E+01	1.045E+02
Sb-127	4.707E+00	6.327E+00	6.873E+00	7.378E+00	8.074E+00
Sb-129	1.157E+01	1.436E+01	1.462E+01	1.463E+01	1.463E+01

Table 2.22-16: Assumptions Used for Locked Rotor Dose Analysis

<u>Source Term</u>	
Core Activity	See Table 2.22-7
Fraction of fuel rods in core assumed to fail for dose considerations (% of core)	8
Centerline melted fuel (%)	1
Radial peaking factor	1.73
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Alkali Metals	12
Fraction of activity released from melted fuel (%)	
Primary to secondary leakage	
Iodine	50
Noble Gas	100
Alkali Metals	100
Iodine chemical form after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
Reactor coolant noble gas activity prior to accident (% fuel defect level)	1.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10

Table 2.22-16 cont'd: Assumptions Used for Locked Rotor Dose Analysis

Release Modeling

SG tube leak rate (gpm total)	1
Steam release to environment (lbm)	
0 – 2 hours	364,000
2 – 8 hours	939,000
> 8 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
RCS mass (lbm)	4.11E5
Secondary Side mass (lbm/per SG)	115,585

Table 2.22-17: Locked Rotor Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr			
I-131	2.017E+01	2.710E+02			
I-132	1.561E+01	7.644E+01			
I-133	3.154E+01	3.755E+02			
I-134	1.349E+01	2.610E+01			
I-135	2.655E+01	2.440E+02			
Cs-134	6.883E+00	9.151E+01			
Cs-136	2.374E+00	2.687E+01			
Cs-137	4.237E+00	5.524E+01			
Rb-86	8.022E-02	1.057E+00			
Kr-85m	1.111E+03	2.954E+03			
Kr-85	7.009E+01	2.793E+02			
Kr-87	1.510E+03	2.239E+03			
Kr-88	2.763E+03	6.080E+03			
Xe-131m	5.414E+01	2.142E+02			
Xe-133m	3.003E+02	1.152E+03			
Xe-133	9.513E+03	3.730E+04			
Xe-135m	3.465E+02	3.480E+02			
Xe-135	2.100E+03	6.771E+03			
Xe-138	1.376E+03	1.380E+03			

Table 2.22-18: Assumptions Used for Single RCCA Withdrawal Analysis

Source Term

Core Activity	See Table 2.22-7
Fraction of fuel rods in core assumed to fail for dose considerations (% of core)	4
Centerline melted fuel (%)	1
Radial peaking factor	1.73
Gap Fractions (% of core activity)	
Iodine	10
Noble Gas	10
Alkali Metals	12
Fraction of activity released from melted fuel (%)	
Primary to secondary leakage	
Iodine	50
Noble Gas	100
Alkali Metals	100
Iodine chemical form after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
Reactor coolant noble gas activity prior to accident (% fuel defect level)	1.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10

Table 2.22-18 cont'd: Assumptions Used for Single RCCA Withdrawal Analysis

Release Modeling	
SG tube leak rate (gpm total)	1
Steam release to environment (lbm)	
0 – 2 hours	364,000
2 – 8 hours	939,000
> 8 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
RCS mass (lbm)	4.11E5
Secondary Side mass (lbm/per SG)	115,585

Table 2.22-19: Single RCCA Withdrawal Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr			
I-131	1.598E+01	2.145E+02			
I-132	1.561E+01	7.644E+01			
I-133	3.154E+01	3.755E+02			
I-134	1.348E+01	2.610E+01			
I-135	2.655E+01	2.440E+02			
Cs-134	5.261E+00	6.934E+01			
Cs-136	1.916E+00	2.065E+01			
Cs-137	3.266E+00	4.197E+01			
Rb-86	6.136E-02	8.012E-01			
Kr-85m	1.111E+03	2.954E+03			
Kr-85	5.573E+01	2.221E+02			
Kr-87	1.510E+03	2.239E+03			
Kr-88	2.763E+03	6.080E+03			
Xe-131m	5.414E+01	2.142E+02			
Xe-133m	3.003E+02	1.152E+03			
Xe-133	9.513E+03	3.730E+04			
Xe-135m	3.465E+02	3.480E+02			
Xe-135	2.100E+03	6.771E+03			
Xe-138	1.376E+03	1.380E+03			

Table 2.22-20: Assumptions Used for Steam Line Break Dose Analysis

Reactor coolant noble gas activity prior to accident (% fuel defect level)	1.0
Reactor coolant iodine activity prior to accident (µCi/gm of DE I-131)	
Pre-accident iodine spike	60
Accident-initiated iodine spike (µCi/gm of DE I-131)	1.0
Reactor coolant iodine appearance rate increase due to the accident-initiated spike (times equilibrium rate)	500
Duration of accident-initiated iodine spike (hr)	5.0
Fraction of fuel rods in core assumed to fail for dose considerations (% of core)	1
Radial peaking factor	1.73
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Alkali Metals	12
SG tube leak rate (gpm total)	1
SG tube leak rate to affected (faulted) SG (gpm)	0.35
SG tube leak rate to unaffected (intact) SGs (gpm)	0.65
Steam release from faulted SG to environment during first two minutes (lbm)	162,000
Time to release initial mass in faulted SG (min)	2
Steam releases from intact SGs (lbm)	
0 – 2 hours	386,000
2 – 8 hours	892,000
> 8 hours	0
Time to cool RCS below 212°F and stop releases from faulted SG (hr)	40
SG iodine water/steam partition coefficient	
Faulted SG	1.0
Intact SGs	0.01

Table 2.22-20 cont'd: Assumptions Used for Steam Line Break Dose Analysis

Iodine chemical form after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
RCS mass (lbm)	4.11E5
Intact Secondary SG Side mass (lbm/per SG)	115,585
Faulted SG mass (lbm)	162,000

Table 2.22-21: Steam Line Break Accident Initiated Iodine Spike Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr	24 hr	40 hr	
I-131	1.423E+01	1.420E+02	5.090E+02	8.488E+02	
I-132	3.430E+01	2.315E+02	2.882E+02	2.886E+02	
I-133	6.487E+01	6.242E+02	1.820E+03	2.509E+03	
I-134	1.097E+01	4.856E+01	4.965E+01	4.965E+01	
I-135	1.924E+01	1.682E+02	3.359E+02	3.666E+02	
Kr-85m	6.751E-01	1.795E+00	2.027E+00	2.047E+00	
Kr-85	4.810E+00	1.917E+01	3.247E+01	4.569E+01	
Kr-87	3.035E-01	4.501E-01	4.521E-01	4.521E-01	
Kr-88	1.149E+00	2.528E+00	2.665E+00	2.667E+00	
Xe-131m	1.544E+00	6.108E+00	1.023E+01	1.416E+01	
Xe-133m	2.177E+00	8.348E+00	1.331E+01	1.732E+01	
Xe-133	1.246E+02	4.884E+02	8.060E+02	1.095E+03	
Xe-135m	3.624E-02	3.640E-02	3.640E-02	3.640E-02	
Xe-135	3.588E+00	1.157E+01	1.496E+01	1.596E+01	
Xe-138	4.869E-02	4.883E-02	4.883E-02	4.883E-02	

Table 2.22-22: Steam Line Break Pre-Accident Iodine Spike Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr	24 hr	40 hr	
I-131	9.797E+00	2.688E+01	6.716E+01	1.045E+02	
I-132	1.211E+01	1.837E+01	1.949E+01	1.950E+01	
I-133	4.069E+01	1.030E+02	2.101E+02	2.717E+02	
I-134	2.307E+00	2.550E+00	2.552E+00	2.552E+00	
I-135	9.956E+00	2.117E+01	3.079E+01	3.255E+01	
Kr-85m	6.751E-01	1.795E+00	2.027E+00	2.047E+00	
Kr-85	4.810E+00	1.917E+01	3.247E+01	4.569E+01	
Kr-87	3.035E-01	4.501E-01	4.521E-01	4.521E-01	
Kr-88	1.149E+00	2.528E+00	2.665E+00	2.667E+00	
Xe-131m	1.544E+00	6.108E+00	1.023E+01	1.416E+01	
Xe-133m	2.177E+00	8.348E+00	1.331E+01	1.732E+01	
Xe-133	1.246E+02	4.884E+02	8.060E+02	1.095E+03	
Xe-135m	3.624E-02	3.640E-02	3.640E-02	3.640E-02	
Xe-135	3.588E+00	1.157E+01	1.496E+01	1.596E+01	
Xe-138	4.869E-02	4.883E-02	4.883E-02	4.883E-02	

Table 2.22-23: Steam Line Break Failed Fuel Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr	24 hr	40 hr	
I-131	1.003E+02	3.950E+02	1.099E+03	1.750E+03	
I-132	7.137E+01	1.390E+02	1.512E+02	1.513E+02	
I-133	1.375E+02	4.638E+02	1.031E+03	1.358E+03	
I-134	6.896E+01	8.668E+01	8.683E+01	8.683E+01	
I-135	1.081E+02	3.225E+02	5.086E+02	5.426E+02	
Cs-134	2.753E+01	1.132E+02	3.261E+02	5.347E+02	
Cs-136	7.669E+00	3.133E+01	8.873E+01	1.431E+02	
Cs-137	1.651E+01	6.789E+01	1.956E+02	3.208E+02	
Rb-86	3.198E-01	1.309E+00	3.727E+00	6.039E+00	
Kr-85m	4.041E+01	1.074E+02	1.213E+02	1.225E+02	
Kr-85	8.447E+00	3.367E+01	5.703E+01	8.024E+01	
Kr-87	5.445E+01	8.075E+01	8.110E+01	8.110E+01	
Kr-88	1.001E+02	2.204E+02	2.324E+02	2.326E+02	
Xe-131m	3.430E+00	1.357E+01	2.273E+01	3.147E+01	
Xe-133m	1.285E+01	4.925E+01	7.854E+01	1.022E+02	
Xe-133	4.604E+02	1.805E+03	2.979E+03	4.047E+03	
Xe-135m	1.245E+01	1.250E+01	1.250E+01	1.250E+01	
Xe-135	7.860E+01	2.534E+02	3.277E+02	3.497E+02	
Xe-138	4.928E+01	4.942E+01	4.942E+01	4.942E+01	

Table 2.22-24: Assumptions Used for Loss of Offsite Power Dose Analysis

Reactor coolant noble gas and alkali metal activity prior to accident (% fuel defect level)	1.0
Reactor coolant iodine activity prior to accident (μCi/gm of DE I-131)	
Pre-accident iodine spike	60
Accident-initiated iodine spike (μCi/gm of DE I-131)	1.0
Reactor coolant iodine appearance rate increase due to the accident-initiated spike (times equilibrium rate)	500
Duration of accident-initiated iodine spike (hr)	5.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10
<u>Release Modeling</u>	
SG tube leak rate (gpm total)	1
Steam release to environment (lbm)	
0 – 2 hours	364,000
2 – 8 hours	939,000
> 8 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
Iodine chemical form after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
RCS mass (lbm)	4.11E5
Secondary Side mass (lbm/per SG)	115,585

Table 2.22-25: Loss of Offsite Power Accident Initiated Iodine Spike Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr			
I-131	1.920E-01	5.291E+00			
I-132	3.536E-01	5.886E+00			
I-133	8.420E-01	2.236E+01			
I-134	8.837E-02	7.218E-01			
I-135	2.311E-01	5.488E+00			
Kr-85m	6.739E-01	1.792E+00			
Kr-85	4.819E+00	1.920E+01			
Kr-87	3.034E-01	4.499E-01			
Kr-88	1.147E+00	2.524E+00			
Xe-131m	1.544E+00	6.110E+00			
Xe-133m	2.177E+00	8.347E+00			
Xe-133	1.244E+02	4.878E+02			
Xe-135m	3.626E-02	3.642E-02			
Xe-135	3.591E+00	1.158E+01			
Xe-138	4.849E-02	4.863E-02			

Table 2.22-26: Loss of Offsite Power Pre-Accident Iodine Spike Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr			
I-131	1.746E-01	1.418E+00			
I-132	1.811E-01	5.778E-01			
I-133	7.131E-01	5.186E+00			
I-134	2.578E-02	3.866E-02			
I-135	1.673E-01	9.462E-01			
Kr-85m	6.739E-01	1.792E+00			
Kr-85	4.819E+00	1.920E+01			
Kr-87	3.034E-01	4.499E-01			
Kr-88	1.147E+00	2.524E+00			
Xe-131m	1.544E+00	6.110E+00			
Xe-133m	2.177E+00	8.347E+00			
Xe-133	1.244E+02	4.878E+02			
Xe-135m	3.626E-02	3.642E-02			
Xe-135	3.591E+00	1.158E+01			
Xe-138	4.849E-02	4.863E-02			

Table 2.22-27: Assumptions Used for Rod Ejection Dose Analysis

Source Term	
Core Activity	See Table 2.22-7
Fraction of fuel rods in core that fail (% of core)	4
Gap Fractions (% of core activity)	
Iodine	10
Noble Gas	10
Alkali Metals	12
Fraction of fuel melting (% of core)	2
Radial peaking factor	1.73
Fraction of activity released from melted fuel (%)	
Containment leakage	
Iodine	25
Noble Gas	100
Alkali Metals	100
Primary to secondary leakage	
Iodine	50
Noble Gas	100
Alkali Metals	100
Reactor coolant noble gas and alkali metal activity prior to accident (% fuel defect level)	1.0
Reactor coolant iodine activity prior to accident (μCi/gm of DE I-131)	1.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10

Table 2.22-27 cont'd: Assumptions Used for Rod Ejection Dose Analysis

<u>Containment Leakage Release Path</u>	
Containment net free volume (ft^3)	2.344E6
Containment leak rates (weight %/day)	
0 – 24 hours	0.1
> 24 hours	0.05
Iodine chemical form in containment (%)	
Elemental	4.85
Organic	0.15
Particulate (cesium iodide)	95
Spray removal in containment	Not Credited
Sedimentation removal in containment (hr^{-1})	
Iodines	Not Credited
Alkali metals	0.2
<u>Primary to secondary Leakage Release Path</u>	
SG tube leak rate (gpm total)	1.0
Steam release to environment (lbm)	
0 – 2 hours	364,000
> 2 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
Iodine chemical form in after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0

Table 2.22-28: Rod Ejection Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	1.210E+02	4.104E+02	1.152E+03	2.579E+03	6.882E+03
I-132	1.293E+02	2.365E+02	2.571E+02	2.572E+02	2.572E+02
I-133	2.399E+02	7.512E+02	1.705E+03	2.321E+03	2.383E+03
I-134	1.304E+02	1.588E+02	1.590E+02	1.590E+02	1.590E+02
I-135	2.077E+02	5.459E+02	8.594E+02	8.952E+02	8.952E+02
Cs-134	5.058E+01	1.111E+02	1.382E+02	1.474E+02	2.262E+02
Cs-136	1.413E+01	3.091E+01	3.828E+01	4.057E+01	5.044E+01
Cs-137	3.034E+01	6.662E+01	8.285E+01	8.842E+01	1.363E+02
Rb-86	5.877E-01	1.287E+00	1.596E+00	1.694E+00	2.209E+00
Kr-85m	1.805E+03	1.904E+03	1.964E+03	1.966E+03	1.966E+03
Kr-85	8.757E+01	9.626E+01	1.194E+02	1.715E+02	6.184E+02
Kr-87	2.458E+03	2.498E+03	2.499E+03	2.499E+03	2.499E+03
Kr-88	4.497E+03	4.676E+03	4.728E+03	4.728E+03	4.728E+03
Xe-131m	8.723E+01	9.581E+01	1.181E+02	1.632E+02	3.471E+02
Xe-133m	4.865E+02	5.324E+02	6.385E+02	7.779E+02	8.676E+02
Xe-133	1.537E+04	1.686E+04	2.061E+04	2.728E+04	4.062E+04
Xe-135m	5.639E+02	5.639E+02	5.639E+02	5.639E+02	5.639E+02
Xe-135	3.411E+03	3.663E+03	3.971E+03	4.037E+03	4.037E+03
Xe-138	2.235E+03	2.235E+03	2.235E+03	2.235E+03	2.235E+03

Table 2.22-29: Assumptions Used for Small Break Loss of Coolant Accident

<u>Source Term</u>	
Core Activity	See Table 2.22-7
Fraction of fuel rods in core that fail (% of core)	100
Gap Fractions (% of core activity)	
I-131	8
Other Iodine	5
Kr-85	10
Other Noble Gas	5
Alkali Metals	12
Fraction of fuel melting (% of core)	2
Radial peaking factor applied to fuel melt	1.73
Fraction of activity released from melted fuel (%)	
Containment leakage	
Iodine	25
Noble Gas	100
Alkali Metals	100
Primary to secondary leakage	
Iodine	50
Noble Gas	100
Alkali Metals	100
Reactor coolant iodine activity (μCi/gm of DE I-131)	1.0
Reactor coolant noble gas and alkali metal activity prior to accident (% fuel defect level)	1.0
Secondary coolant iodine activity prior to accident (μCi/gm of DE I-131)	0.1
Secondary coolant alkali metal activity prior to accident (% of primary concentration)	10

Table 2.22-29 cont'd: Assumptions Used for the SBLOCA Analysis

<u>Containment Leakage Release Path</u>	
Containment net free volume (ft^3)	2.344E6
Containment sprayed volume (ft^3)	2.014E6
Fan cooler units	
Number in operation	2
Flow rate (per unit)	31,250
Containment leak rates (weight %/day)	
0 – 24 hours	0.1
> 24 hours	0.05
Iodine chemical form in containment (%)	
Elemental	4.85
Organic	0.15
Particulate (cesium iodide)	95
Spray Operation	
Time to initiate sprays	30.0 minutes
Time to terminate spray operation	1.0 hour
Spray flow rates (gpm)	1730
Spray fall height (ft)	125
Removal Coefficients (hr^{-1})	
Spray elemental iodine removal	20.0
Spray particulate removal	3.94
Sedimentation particulate removal (after spray termination)	0.2
<u>Primary to secondary Leakage Release Path</u>	
SG tube leak rate (gpm total)	1.0
Steam release to environment (lbm/sec)	
0 – 2 hours	364,000
> 2 hours	0
SG iodine and alkali metal water/steam partition coefficient	0.01
Iodine chemical form after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0

Table 2.22-30: SBLOCA Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	3.655E+02	5.272E+02	6.056E+02	6.522E+02	7.929E+02
I-132	2.830E+02	3.292E+02	3.318E+02	3.318E+02	3.318E+02
I-133	4.892E+02	6.813E+02	7.539E+02	7.672E+02	7.685E+02
I-134	8.401E+01	8.544E+01	8.545E+01	8.545E+01	8.545E+01
I-135	4.317E+02	5.639E+02	5.924E+02	5.932E+02	5.932E+02
Cs-134	1.201E+02	1.744E+02	1.968E+02	2.036E+02	2.617E+02
Cs-136	3.349E+01	4.850E+01	5.460E+01	5.628E+01	6.355E+01
Cs-137	7.197E+01	1.045E+02	1.180E+02	1.221E+02	1.574E+02
Rb-86	1.395E+00	2.022E+00	2.278E+00	2.351E+00	2.730E+00
Kr-85m	3.935E+03	4.152E+03	4.282E+03	4.288E+03	4.288E+03
Kr-85	2.892E+02	3.179E+02	3.944E+02	5.664E+02	2.043E+03
Kr-87	5.362E+03	5.448E+03	5.451E+03	5.451E+03	5.451E+03
Kr-88	9.804E+03	1.019E+04	1.031E+04	1.031E+04	1.031E+04
Xe-131m	1.883E+02	2.069E+02	2.550E+02	3.523E+02	7.494E+02
Xe-133m	1.058E+03	1.158E+03	1.389E+03	1.692E+03	1.887E+03
Xe-133	3.338E+04	3.662E+04	4.476E+04	5.924E+04	8.818E+04
Xe-135m	1.229E+03	1.229E+03	1.229E+03	1.229E+03	1.229E+03
Xe-135	7.431E+03	7.981E+03	8.652E+03	8.794E+03	8.795E+03
Xe-138	4.873E+03	4.874E+03	4.874E+03	4.874E+03	4.874E+03

Table 2.22-31: Assumptions Used for FHA in Containment Dose Analysis

Radial peaking factor	1.73
Fuel damaged (number of assemblies)	1
Time from shutdown before fuel movement (hr)	100
Activity in the damaged fuel assembly (Ci)	
I-131	6.06E5
I-133	6.38E4
I-135	4.68E1
Kr-85	8.82E3
Xe-131m	7.61E3
Xe-133m	1.49E4
Xe-133	9.97E5
Xe-135	2.03E2
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Water depth	22 feet
Overall pool iodine scrubbing factor	200
Iodine chemical form in release to atmosphere (%)	
Elemental	70
Organic	30
Particulate	0
Filter efficiency	No filtration assumed
Isolation of release	No isolation assumed
Time to release all activity (hours)	2

Table 2.22-32: FHA in Containment – Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2.0 hr				
I-131	2.420E2				
I-133	1.592E1				
I-135	1.168E-2				
Kr-85	8.820E2				
Xe-131m	3.805E2				
Xe-133m	7.450E2				
Xe-133	4.985E4				
Xe-135	1.015E1				

Table 2.22-33: Assumptions Used for FHA in the Fuel Handling Building Dose Analysis

Radial peaking factor (PWR fuel)	1.73
(BWR fuel)	1.5
Fuel damaged (number of assemblies)	1.2 PWR (314 rods) + 52 BWR
Time from shutdown before fuel movement (PWR) (hr)	100
(BWR fuel) (yr)	4
Activity in the damaged fuel assemblies (Ci)	
I-131	7.21E5
I-133	7.59E4
I-135	5.57E1
Kr-85	1.41E5
Xe-131m	9.06E3
Xe-133m	1.77E4
Xe-133	1.19E6
Xe-135	2.41E2
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Water depth	21 feet
Overall pool iodine scrubbing factor	200
Iodine chemical form in release to atmosphere (%)	
Elemental	70
Organic	30
Particulate	0
Spent Fuel Pool Ventilation System Filter efficiency	
Elemental	95
Organic	95
Particulate	95
Isolation of release	No isolation assumed
Time to release all activity (hours)	2

Table 2.22-34: FHA in Fuel Handling Building - Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2.0 hr				
I-131	1.439E1				
I-133	9.471E-1				
I-135	6.950E-4				
Kr-85	1.410E4				
Xe-131m	4.530E2				
Xe-133m	8.850E2				
Xe-133	5.950E4				
Xe-135	1.205E1				

Table 2.22-35: Assumptions Used for Letdown Line Break Analysis

Reactor coolant noble gas activity prior to accident (% fuel defect level)	1.0
Reactor coolant iodine activity prior to accident (μCi/gm of DE I-131)	1.0
Reactor coolant iodine appearance rate increase due to the accident-initiated spike (times equilibrium rate)	500
Letdown line break flow (gpm)	200
Duration of letdown line break (minutes)	30
Break flow flashing fraction	0.4

Table 2.22-36: Letdown Line Break Integrated Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	0.5 hr				
I-131	1.391E+02				
I-132	4.483E+02				
I-133	6.637E+02				
I-134	1.842E+02				
I-135	2.144E+02				
Kr-85m	3.576E+01				
Kr-85	2.275E+02				
Kr-87	2.071E+01				
Kr-88	6.487E+01				
Xe-131m	7.314E+01				
Xe-133m	1.040E+02				
Xe-133	5.916E+03				
Xe-135m	5.183E+00				
Xe-135	1.795E+02				
Xe-138	7.197E+00				

Table 2.22-37: Assumptions Used for Waste Gas Decay Tank Rupture Dose Analysis

Gas Decay Tank Rupture Source Term

Gas decay tank inventory (Ci)	
Kr-83m	19.1
Kr-85m	138.0
Kr-85	4100.0
Kr-87	46.0
Kr-88	172.0
Xe-131m	775.0
Xe-133m	903.0
Xe-133	58500.0
Xe-135m	56.6
Xe-135	900.0
Xe-138	5.16

Release modeling

Time to release all GDT activity (hours)	2

Table 2.22-38: WGDTR Integrated Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr				
Kr-83m	1.910E+01				
Kr-85m	1.380E+02				
Kr-85	4.100E+03				
Kr-87	4.600E+01				
Kr-88	1.720E+02				
Xe-131m	7.750E+02				
Xe-133m	9.030E+02				
Xe-133	5.850E+04				
Xe-135m	5.660E+01				
Xe-135	9.000E+02				
Xe-138	5.160E+00				

6.3.3 Radiological Consequences Analysis

6.3.3.1 Introduction

The evaluation of the radiological consequences of a steam generator tube rupture (SGTR) assumes that the reactor has been operating at the Technical Specification limits for primary coolant activity and primary to secondary leakage for sufficient time to establish equilibrium concentrations of radio-nuclides in the reactor coolant and in the secondary coolant. Radio-nuclides from the primary coolant enter the steam generator, via the ruptured tube and primary to secondary leakage, and are released to the atmosphere through the steam generator safety or power operated relief valves (PORVs) and via the condenser air ejector exhaust.

The quantity of radioactivity released to the environment, due to a SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow, break flow flashing, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the generator and liquid-vapor partitioning in the turbine condenser hot well. All of these parameters were conservatively evaluated for a design basis double ended rupture of a single tube.

The most recent SGTR radiological consequences analysis performed by Westinghouse for HNP, documented in WCAP-12403 and Supplement 1 to WCAP-12403 (References 1 and 2) were performed using the analysis methodology developed in Supplement 1 to WCAP-10698 (Reference 3). The methodology was developed by the SGTR Subgroup of the Westinghouse Owners Group (WOG) and was approved by the Nuclear Regulatory Commission (NRC) in a Safety Evaluation Report (SER) dated December 17, 1985. The SGTR radiological consequences analysis was performed in support of the HNP model Δ75 replacement steam generator program using this methodology with some variations. These variations in methodology reflect the latest accepted methods and are identified in this report.

Section 6.3.2 of this report presents the mass releases for the SGTR event assuming failure and isolation of the ruptured steam generator PORV for analyses modeling the model Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt. The resulting offsite and control room doses are calculated in this section.

This section includes the methods and assumptions used to analyze the radiological consequences of the SGTR event, as well as the calculated results.

6.3.3.2 Input Parameters and Assumptions

The input data for the SGTR radiological consequences analysis is documented in Reference 4. Major assumptions and parameters are summarized in Table 6.3.3-1.

The total effective dose equivalent (TEDE) doses are determined at the exclusion area boundary (EAB) for the worst 2-hour interval. The TEDE dose at the low population zone (LPZ) and for the control room (CR) personnel are determined for the duration of the event. The interval for determining control room doses extends beyond the time when the releases are terminated. This accounts for the additional dose to the operators in the control room, which will continue for as long as the activity is circulating within the control room envelope.

The TEDE dose is equivalent to the committed effective dose equivalent (CEDE) dose or inhalation dose plus the effective dose equivalent dose (EDE) dose for the duration of exposure to the cloud. The dose conversion factors (DCFs) used in determining the CEDE dose are from Reference 13 and are given in Table 6.3.3-6. The dose conversion factors used in determining the EDE dose are from Reference 15 and are listed in Table 6.3.3-6.

6.3.3.2.1 Source Term Assumptions

The radio-nuclide concentrations in the primary and secondary system, prior to and following the SGTR, are determined as follows.

1. The iodine concentrations in the reactor coolant are based upon pre-accident and accident-initiated iodine spikes as outlined in Regulatory Guide (RG) 1.183, Appendix F (Reference 5).

    a. Pre-accident Spike - A reactor transient has occurred prior to the SGTR and has raised the primary coolant iodine concentration to 60 μCi/gm of Dose Equivalent (D.E.) I-131.

    b. Accident-Initiated Spike - The primary coolant iodine concentration is initially at the Technical Specification limit, specified in μCi/gm of D.E. I-131. Following the primary system depressurization and reactor trip associated with the SGTR, an iodine spike is initiated in the primary system. This spike increases the iodine release rate from the fuel to the coolant to a value 335 times greater than the release rate corresponding to the initial primary system iodine concentration. This release rate (the equilibrium iodine appearance rate) is calculated to match the rate of iodine removal from the RCS. Iodine removal from the RCS is the combination of decay, leakage and cleanup.

2. The initial secondary coolant iodine concentration is 0.1 μCi/gm of D.E. I-131.

3. The chemical form of iodine in the primary and secondary coolant is assumed to be 97% elemental and 3% organic.

4. The initial concentration of noble gases in the reactor coolant is based on one percent defective fuel, which corresponds to the Technical Specification limit of 100/E-bar.

5. No noble gases are present in the secondary system at the start of the event.

The concentration of iodine and noble gas nuclides in the reactor coolant system (RCS) has been calculated based on a one percent fuel defect level for the SGR/Uprating program. The concentration data presented in Table 6.3.3-2 is used in the SGTR analysis.

The conversion from the one percent fuel defect values in Table 6.3.3-2 to DE I-131 employs dose conversion factors (DCFs). In the Reference 1 and 2 analyses the thyroid dose conversion factors are from Regulatory Guide (RG) 1.109 (Reference 6). In order to be consistent with current analysis techniques and NRC expectations thyroid dose conversion factors from International Commission on Radiological Protection (ICRP)-30 (Reference 7) are used in this analysis. These DCFs are used in the calculation of the initial RCS iodine concentrations. The ICRP-30 thyroid dose conversion factors used in the analysis are presented in Table 6.3.3-3.

The iodine spike model used in the Reference 1 and 2 analyses calculated equilibrium iodine appearance rates based on a letdown flow of 60 gpm with 90 percent cleanup and used a spike appearance rate of 500 times the equilibrium appearance rate.

The Nuclear Safety Advisory Letter (NSAL) in Reference 8 identified non-conservative assumptions that have been used in the calculation of accident-initiated iodine spiking rates in the primary coolant. The conservative spike model calculates the equilibrium iodine appearance rates based on a letdown flow of 120 gpm with perfect cleanup. This flow is conservatively increased by 10 percent to cover uncertainties in the flow. In addition, a total of 42 gpm leakage from the RCS allowed by the Technical Specifications (which also remove iodine from the RCS) is considered in the calculations. The effective letdown flow increases from 54 gpm with the Reference 1 and 2 spike model to 174 gpm with the spike model suggested by the NSAL. The 174 gpm is the total of 120 gpm letdown flow with perfect cleanup increased by 10 percent to 132 gpm (to cover uncertainty), 10 gpm identified leakage from the RCS, 1 gpm unidentified leakage from the RCS, and 31 gpm controlled leakage. (Although inclusion of the controlled leakage in the effective letdown flow is conservative, it is not necessary since this flow does not remove activity from the RCS.)

The spike is allowed to continue until 8 hours from the start of the event. This bounds the time calculated for all iodine initially contained in the gap of the defective fuel to be transferred to the coolant at the spike appearance rate being modeled. In the Reference 1 and 2 analyses, the spike was assumed to be terminated at 2.78 hours. The spike duration was extended in response to NRC comments on recent analyses performed for other plants. This has little impact on the SGTR analysis, since the majority of the iodine releases end shortly after the ruptured steam generator PORV is isolated at about 30 minutes from the start of the event.

The initial RCS iodine activities used in the analysis are presented in Table 6.3.3-4.
The iodine appearance rates used in the analysis are presented in Table 6.3.3-5.

6.3.3.2.2 Dose Calculation Assumptions

Offsite power is assumed to be lost at reactor trip. This assumption was used in the thermal-hydraulic analysis (Section 6.3.2) to maximize break flow and steam release though the ruptured steam generator PORV. Prior to reactor trip, a condenser iodine partition factor of 0.01 is assumed. After reactor trip and loss of offsite power, flow to the condenser is isolated. This condenser iodine partition factor is consistent with the RG 1.183 (Reference 5) steam/water partition coefficient for SGs.

The iodine transport model used in this analysis accounts for break flow flashing, steaming, and partitioning. The model assumes that a fraction of the iodine carried by the break flow becomes airborne immediately due to flashing and atomization. Droplet removal by the dryers is conservatively neglected. The fraction of primary coolant iodine that is not assumed to become airborne immediately mixes with the secondary water and is assumed to become airborne at a rate proportional to the steaming rate. The 0.01 steam/water partition coefficient from RG 1.183 (Reference 5) is used.

In the iodine transport model, the time dependent iodine removal efficiency for scrubbing of steam bubbles as they rise from the rupture site to the water surface was not calculated and was conservatively neglected. Although this removal was calculated and credited in the Reference 1 and 2 analyses using a model based on that proposed in NUREG-0409 (Reference 9), it is no longer considered in standard Westinghouse analyses.

All of the iodine in the flashed break flow is assumed to be transferred instantly out of the steam generator to the atmosphere.

The issue of tube bundle uncovery was considered in a Westinghouse Owners Group (WOG) program (Reference 10). The WOG program concluded that the effect of tube uncovery is essentially negligible for the limiting SGTR transient. The WOG program concluded that the steam generator tube uncovery issue could be closed without any further investigation or generic restrictions. The NRC review of the WOG submittal (Reference 11) concluded "... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue resolved." This modeling is different from that used in the Reference 1 and 2 analyses. Those analyses were completed prior to the resolution of the tube uncovery issue and conservatively modeled the direct release of all iodine transferred to the ruptured steam generator in the break flow when the tubes were assumed to be uncovered.

Since there is no penalty taken for tube uncovery and no iodine scrubbing is credited, the location of the tube rupture is not significant for the radiological analysis. The thermal and hydraulic analysis presented in Section 6.3.2 has conservatively addressed the issue of the location of the tube rupture in the calculations of break flow and flashing of break flow.

No credit is taken for the radioactive decay during release and transport, or for cloud depletion by ground deposition during transport to the control room, exclusion area boundary (EAB) or outer boundary of the low population zone (LPZ).

All noble gases in the break flow and primary-to-secondary leakage are assumed to be transferred instantly out of the steam generator to the atmosphere.

Iodine and noble gas decay constants are presented in Table 6.3.3-7. These decay constants were calculated from half-lives given in Reference 12.

Short-term atmospheric dispersion factors (χ/Qs) for accident analysis and breathing rates are provided in Table 6.3.3-8. The offsite and control room breathing rates and control room occupancy factors are consistent with RG 1.183.

Offsite Dose Calculation Model

The TEDE dose is calculated for the worst 2 hour period at the EAB. At the LPZ the TEDE dose is calculated up to the time all releases are terminated, which is the RHR cut in time used in the thermal and hydraulic analysis. The TEDE doses are obtained by combining the CEDE doses and the EDE doses.

Offsite inhalation doses (CEDE) are calculated using the following equation.

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} (BR)_j (\chi/Q)_j \right) \right]$$

where:

D_{CEDE} = CEDE dose via inhalation (rem).

DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci) (Table 6.3.3-6)

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)
$(BR)j$ = breathing rate during time interval j (m^3/sec) (Table 6.3.3-8)
$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 6.3.3-8)

Offsite external exposure (EDE) doses are calculated using the following equation:

$$D_{EDE} = \sum_i \left[DCF_i \left(\sum_j (IAR)_{ij} (\chi/Q)_j \right) \right]$$

where:

D_{EDE} = external exposure dose via cloud immersion (rem)
DCF_i = EDE dose conversion factor via external exposure for isotope i (rem·m^3/Ci·sec) (Table 6.3.3-6)
$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)
$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 6.3.3-8)

Control Room Dose Calculation Models

CEDE (doses due to inhalation) and EDE (doses due to external exposure) are calculated for 30 days in the control room. Although all releases are terminated when the RHR system is put in service, the calculation is continued to account for additional doses due to continued occupancy.

The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake are used to determine the activity available at the control room intake. The inflow (filtered and unfiltered) to the control room and the control room filtered recirculation flow are used to calculate the concentration of activity in the control room. Control room parameters used in the analysis are presented in Table 6.3.3-9.

Control room inhalation doses are calculated using the following equation:

$$D_{CEDE} = \sum_i \left[DCF_i \left(\sum_j Conc_{ij} * (BR)_j \right) \right]$$

where:

D_{CEDE} = CEDE dose via inhalation (rem)
DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci) (Table 6.3.3-6)
$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow (Ci-sec/m^3)
$(BR)j$ = breathing rate during time interval j (m^3/sec) (Table 6.3.3-8)

Control room external exposure doses are calculated using the following equation:

$$D_{EDE} = \left(\frac{1}{GF} \right) * \sum_i DCF_i \left(\sum_j Conc_{ij} \right)$$

where:

D_{EDE} = external exposure dose via cloud immersion in rem.
GF = geometry factor, calculated based on Reference 14, using the equation

$$GF = \frac{1173}{V^{0.338}}$$ where V is the control room volume in ft^3

DCF_i = EDE dose conversion factor via external exposure for isotope i ($rem \cdot m^3/Ci \cdot sec$) (Table 6.3.3-6)

$Conc_{ij}$ = concentration in the control room of isotope i, during time interval j, calculated dependent upon inleakage, filtered recirculation and filtered inflow ($Ci\text{-}sec/m^3$)

The control room HVAC begins in normal mode. Once the safety injection actuation setpoint is reached at ~178 seconds and after a delay of 15 seconds the control room HVAC is switched to the post-accident recirculation mode. After 2 hours of operation in post-accident recirculation mode the operator switches the control room HVAC system to the pressurized mode.

6.3.3.2.3 Mass Transfer Assumptions

Break flow, flashing break flow and steam releases from the intact and ruptured steam generators are modeled using data from the thermal and hydraulic analysis in Section 6.6.2 of this report.

A total primary to secondary leak rate is assumed to be 1.0 gpm. The leak is assumed to be distributed with 0.7 gpm to the two intact steam generators and 0.3 gpm to the ruptured steam generator. The leakage to the intact steam generators is assumed to persist for the duration of the accident. This modeling is consistent with the Reference 1 and 2 analyses. Atmospheric conditions are assumed in determining the density for this leakage.

In addition to the releases calculated in the thermal hydraulic analysis presented in Section 6.3.2, steam released from the ruptured steam generator to the turbine driven auxiliary feedwater (TDAFW) pump is considered in the dose analysis. A flow of 41,310 lbm/hr is considered from the time of auxiliary feedwater initiation until the ruptured steam generator is isolated. The iodine contained in this steam, determined from the steam generator activity and the water/steam partition coefficient of 100, is assumed to be released directly to the atmosphere. This flow was not modeled in the Reference 1 and 2 analyses and, since it is assumed to be released directly to the atmosphere, is conservative.

6.3.3.3 Description of Analyses and Evaluations

Offsite and control room doses are calculated for the limiting thermal hydraulic analysis presented in Section 6.3.2 of this report. The limiting thermal hydraulic analysis corresponds to the analysis performed at the uprated NSSS power of 2912.4 MWt with Model Δ75 replacement steam generators. For this case the mass transfer data is taken from Table 6.3.2-2, and Figures 6.3.2-6, 6.3.2-8, 6.3.2-9 and 6.3.2-10.

6.3.3.4 Acceptance Criteria

The doses at the exclusion area boundary (EAB) and the LPZ for an SGTR with an assumed pre-accident iodine spike must be within the RG 1.183 limit of 25 rem TEDE. The doses at the EAB and the LPZ for an SGTR with an assumed accident-initiated iodine spike must be within the RG 1.183 limit of 2.5 rem TEDE. The doses in the control room must be less than the 10CFR50.67 dose limit of 5 rem TEDE.

The exclusion area boundary doses are calculated for the worst 2 hours. The LPZ doses are calculated up to the time all releases are terminated, which is the Residual Heat Removal (RHR) cut in time (8 hours) used in the thermal and hydraulic analysis in Section 6.3.2. The control room doses are calculated for 30 days.

6.3.3.5 Results

The pre-accident iodine spike TEDE doses for the SGTR analysis with Model Δ75 replacement steam generators at the uprated NSSS power of 2912.4 MWt are tabulated in Table 6.3.3-12. The table includes the applicable limit. The applicable limits are met. Table 6.3.3-14 presents the pre-accident iodine spike integrated activity released to the atmosphere.

Table 6.3.3-13 presents the accident-initiated iodine spike TEDE doses calculated based on a primary coolant iodine limit of 1.0 μCi/gm D.E. I-131, and spike appearance rates calculated with conservative assumptions consistent with those suggested in Reference 8. The results in the table demonstrate that the applicable limits are met. Table 6.3.3-15 presents the accident-initiated iodine spike integrated activity released to the atmosphere.

6.3.3.6 Conclusions

The potential radiological consequences of a steam generator tube rupture were evaluated for HNP in support of the SGR/Uprating program. Since it was concluded in Section 6.3.1 that steam generator overfill will not occur for a design basis SGTR, an analysis was performed to determine the offsite radiation doses assuming the limiting single failure for offsite doses. The thermal hydraulic results from this analysis are presented in Section 6.3.2. The resulting doses at the exclusion area boundary, low population zone, and control room (presented in Section 6.3.3) are within the allowable guidelines.

6.3.3.7 References

1. WCAP-12403, "LOFTTR2 Analysis for a Steam Generator Tube Rupture with Revised Operator Action Times for Shearon Harris Nuclear Power Plant," Huang, Lewis, Marmo, Rubin, November 1989.
2. Supplement 1 to WCAP-12403, "Steam Generator Tube Rupture Analysis for Shearon Harris Nuclear Power Plant," Lewis, Lowe, Monahan, Rubin, Tanz, November 1992.
3. Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Lewis, Huang, Rubin, March 1986.
4. Westinghouse Letter CQL-01-024, "Alternate Source Term Radiological Consequences Input Assumptions," dated 03/24/01. (Accepted via Letter HW/01-024, "Harris Plant Alternate Source Term Owners Review of Inputs," 03/21/01).
5. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.
6. Regulatory Guide 1.109, Rev. 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," US Nuclear Regulatory Commission, October 1977.
7. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers", ICRP Publication 30, Volume 3 No. 1-4, 1979.

8. Nuclear Safety Advisory Letter, NSAL-00-004, "Nonconservatisms in Iodine Spiking Calculations", March 2000.

9. NUREG-0409 "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture," A. K. Postma, P. S. Tam.

10. WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992.

11. Letter from Robert C. Jones to Lawrence A. Walsh, "Westinghouse Owners Group-Steam Generator Tube Uncovery Issue", March 10, 1993.

12. ENDF-223, "ENDF/B-IV Fission-Product Files: Summary of Major Nuclide Data," T. R. England and R. E. Schenter, October 1975.

13. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88—020, September 1988.

14. K. G. Murphy and K. W. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," published in Proceedings of 13th AEC Air Cleaning Conference, Atomic Energy Commission (now NRC), August 1974.

15. EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water and Soil," EPA 402-R-93-081, September 1993.

Table 6.3.3-1

Summary of Parameters Used in Evaluating
the Radiological Consequences of
a Steam Generator Tube Rupture

I. Source Data	
A. Core power level, MW_t	2900
B. Reactor coolant iodine activity:	
1. Accident-Initiated Spike	The initial RC iodine activities are presented in Table 6.3.3-4. The iodine appearance rates assumed for the accident-initiated spike are presented in Table 6.3.3-5.
2. Pre-Accident Spike	Primary coolant iodine activities based on 60 μCi/gm of D.E. I-131 are presented in Table 6.3.3-4.
C. Noble Gas Activity	Primary coolant noble gas activities based on 1 percent fuel defects are presented in Table 6.3.3-2. No noble gases are contained in the secondary system.
D. Secondary system initial activity	Dose equivalent of 0.1 μCi/gm of I-131, presented in Table 6.3.3-4.
E. Reactor coolant initial mass, grams	1.73×10^8
F. Steam generator initial mass (each), grams	4.34×10^7
G. Offsite power	Lost at time of reactor trip
H. Primary-to-secondary leakage duration for intact SG, hours	8
I. Species of iodine	97 percent elemental, 3 percent organic

Table 6.3.3-1
(Continued)

II. Activity Release Data	
A. Ruptured steam generator	
1. Rupture flow	See Table 6.3.2-2 & Figure 6.3.2-6
2. Flashed rupture flow	See Table 6.3.2-2 & Figure 6.3.2-8
3. Steam releases	See Table 6.3.2-2 & Figure 6.3.2-9 An additional 41,310 lbm/hr to TDAFW pump is modeled until ruptured SG isolation.
4. Iodine partition factor for rupture flow	
Non-flashed	100
Flashed	1.0
B. Intact steam generators	
1. Primary-to-secondary leakage, gpm	0.7
2. Steam releases	See Table 6.3.2-2 & Figure 6.3.2-10
3. Iodine partition factor	100
C. Condenser	
1. Iodine partition factor	100
D. Atmospheric Dispersion Factors	See Table 6.3.3-8

Table 6.3.3-2

Reactor Coolant Fission Product Specific Activity Based on 1 Percent Fuel Defects

Nuclide	Specific Activity (μCi/gm)
I-131	1.71E+00
I-132	2.47E+00
I-133	7.23E+00
I-134	5.67E-01
I-135	1.84E+00
Kr-85m	1.73
Kr-85	10.6
Kr-87	1.10
Kr-88	3.21
Xe-131m	3.41
Xe-133m	4.86
Xe-133	276.
Xe-135m	0.436
Xe-135	8.52
Xe-138	0.63

Table 6.3.3-3

Thyroid Dose Conversion Factors (Reference 7*)

Nuclide	DCF (Rem/Curie)
I-131	1.07×10^6
I-132	6.29×10^3
I-133	1.81×10^5
I-134	1.07×10^3
I-135	3.14×10^4

* Reference 7 provides the dose conversion factors in units of sievert/becquerel.

Table 6.3.3-4

Iodine Specific Activities (μCi/gm) in the
Primary Coolant Based on 1.0 and 60.0 μCi/gm of D.E. I-131 and in the
Secondary Coolant Based on 0.1 μCi/gm of D.E. I-131

	Primary Coolant		*Secondary Coolant*
Nuclide	1 μCi/gm	60 μCi/gm	0.1 μCi/gm
1-131	0.570	34.20	0.0570
1-132	0.823	49.38	0.0823
1-133	2.408	144.48	0.2408
1-134	0.189	11.34	0.0189
1-135	0.613	36.78	0.0613

Table 6.3.3-5

Iodine Spike Appearance Rates (Curies/Minute)
Based on 1.0 μCi/gm of D.E. I-131 Primary Coolant Activity
Calculated with Assumptions as Explained in NSAL-00-004 (Reference 8)

Primary Activity	I-131	I-132	I-133	I-134	I-135
1.0 μCi/gm D.E. I-131	127.6	422.4	608.4	186.3	197.3

Table 6.3.3-6

CEDE (Reference 13) and EDE (Reference 15) Dose Conversion Factors

Nuclide	CEDE DCF (rem/Ci)	EDE DCF (rem·m^3/Ci·sec)
I-131	3.29E4	6.734E-2
I-132	3.81E2	0.4144
I-133	5.85E3	0.1088
I-134	1.31E2	0.4810
I-135	1.23E3	0.2953
Kr-85m	N/A	2.768E-2
Kr-85	N/A	4.403E-4
Kr-87	N/A	0.1524
Kr-88	N/A	0.3774
Xe-131m	N/A	1.439E-3
Xe-133m	N/A	5.069E-3
Xe-133	N/A	5.772E-3
Xe-135m	N/A	7.548E-2
Xe-135	N/A	4.403E-2
Xe-138	N/A	0.2135

Table 6.3.3-7

Decay Constants (Reference 12)

Nuclide	Decay Constant (1/hr)
I-131	0.00359
I-132	0.303
I-133	0.0333
I-134	0.791
I-135	0.105
Kr-85m	0.155
Kr-85	7.37E-6
Kr-87	0.547
Kr-88	0.248
Xe-131m	0.00241
Xe-133m	0.0130
Xe-133	0.00546
Xe-135m	2.72
Xe-135	0.0756
Xe-138	2.93

Table 6.3.3-8

Atmospheric Dispersion Factors and Breathing Rates

Time (hours)	Exclusion Area Boundary χ/Q (sec/m^3)	Low Population Zone χ/Q (sec/m^3)	Control Room χ/Q (sec/m^3)	Offsite Breathing Rate (m^3/sec)	Control Room Breathing Rate (m^3/sec)	Control Room Occupancy Factor*
0 - 2	6.17 x 10^{-4}	1.4 x 10-4	4.08 x 10-3	3.5 x 10-4	3.5 x 10-4	1.0
2 - 8	----	1.4 x 10-4	4.08 x 10-3	3.5 x 10-4	3.5 x 10-4	1.0
8 - 24	----	----	1.16 x 10-3	1.8 x 10-4	3.5 x 10-4	1.0
24 - 96	----	----	3.25 x 10-4	2.3 x 10-4	3.5 x 10-4	0.6
>96	----	----	1.23 x 10-5	2.3 x 10-4	3.5 x 10-4	0.4

*These occupancy factors (from Reference 5) have been conservatively incorporated in the atmospheric dispersion factors. This is conservative since it does not allow the benefit of reduced occupancy for activity already present in the control room from earlier periods.

Table 6.3.3-9

Control Room Model

Control Room Isolation Signal Generated	Time of SI signal from Section 6.3.2
Delay in Control Room Isolation After Isolation Signal is Generated	30 Seconds
Control Room Volume	71000 ft^3
Control Room Unfiltered In-Leakage	500 cfm
Control Room Unfiltered Inflow	
Normal Mode	1050 cfm
Post Accident Recirculation Mode	0 cfm
Post Accident Pressurization Mode	0 cfm
Control Room Filtered Inflow	
Normal Mode	0 cfm
Post Accident Recirculation Mode	0 cfm
Post Accident Pressurization Mode	400 cfm
Control Room Filtered Recirculation	
Normal Mode	0 cfm
Post Accident Recirculation Mode	400 cfm
Post Accident Pressurization Mode	3600 cfm
Control Room Filter Efficiency	
Elemental	99%
Organic	99%
Particulate	99%
Operator Action Time to Switch to Pressurization Mode	2 hours

Table 6.3.3-10

Not Used

Table 6.3.3-11

Not Used

Table 6.3.3-12

Pre-Accident Iodine Spike TEDE Doses

	RSG and Uprated Doses (Rem)	Allowable Guideline Value
Pre-Accident Iodine Spike - TEDE		
Exclusion Area Boundary (0-2 hr.)	2.20	25
Low Population Zone (0-8 hr.)	0.60	25
Control Room (0-30 Days)	1.60	5

Table 6.3.3-13

Accident-Initiated Iodine Spike Thyroid Doses
Iodine Spike Appearance Rates
Based on 1.0 μCi/gm of D.E. I-131 Primary Coolant Activity
Calculated with Conservative Assumptions as Explained in NSAL-00-004 (Reference 8)

	RSG and Uprated Doses (Rem)	Allowable Guideline Value
Accident-Initiated Iodine Spike - TEDE		
Exclusion Area Boundary (0-2 hr.)	1.30	2.5
Low Population Zone (0-8 hr.)	0.40	2.5
Control Room (0-30 Days)	0.90	5

Table 6.3.3-14

Pre-Accident Iodine Spike Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr			
I-131	1.443E+02	1.529E+02			
I-132	2.009E+02	2.078E+02			
I-133	6.035E+02	6.378E+02			
I-134	3.634E+01	3.648E+01			
I-135	1.507E+02	1.591E+02			
Kr-85m	9.709E+01	9.758E+01			
Kr-85	6.548E+02	6.613E+02			
Kr-87	5.216E+01	5.223E+01			
Kr-88	1.746E+02	1.752E+02			
Xe-131m	2.098E+02	2.118E+02			
Xe-133m	3.007E+02	3.035E+02			
Xe-133	1.703E+04	1.719E+04			
Xe-135m	8.687E+00	8.687E+00			
Xe-135	5.053E+02	5.089E+02			
Xe-138	1.229E+01	1.229E+01			

Table 6.3.3-15

Accident-Initiated Iodine Spike Activity Released to Atmosphere

	Integrated Activity (Ci) Released Until End of Time Period				
Nuclide	2 hr	8 hr			
I-131	7.165E+01	8.159E+01			
I-132	2.182E+02	2.283E+02			
I-133	3.381E+02	3.798E+02			
I-134	8.667E+01	8.776E+01			
I-135	1.073E+02	1.174E+02			
Kr-85m	9.709E+01	9.758E+01			
Kr-85	6.548E+02	6.613E+02			
Kr-87	5.216E+01	5.223E+01			
Kr-88	1.746E+02	1.752E+02			
Xe-131m	2.098E+02	2.118E+02			
Xe-133m	3.007E+02	3.035E+02			
Xe-133	1.703E+04	1.719E+04			
Xe-135m	8.687E+00	8.687E+00			
Xe-135	5.053E+02	5.089E+02			
Xe-138	1.229E+01	1.229E+01			