NMC
Committed to Nuclear Excellence

Kewaunee Nuclear Power Plant
N490 Highway 42
Kewaunee, WI 54216-9511
920.388.2560

Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241
920.755.2321

Kewaunee / Point Beach Nuclear
Operated by Nuclear Management Company, LLC

NRC-02-024

March 19, 2002

10 CFR 50.90
10 CFR 50.67

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Ladies/Gentlemen:

Docket 50-305
Operating License DPR-43
Kewaunee Nuclear Power Plant
Revision to the Design Basis Radiological Analysis Accident Source Term

References: 1) Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," dated July 2000

2) NUREG 0800, Standard Review Plan, SRP-15.0.1, Rev. 0 "Radiological Consequence Analyses Using Alternative Source Terms," dated July 2000

Nuclear Management Company, LLC, (NMC) proposes, in accordance with provisions of 10 CFR 50.67, "Accident Source Term," to revise the Kewaunee Nuclear Power Plant (KNPP) accident source term used for design basis radiological analyses. This revision follows guidance set forth by the Nuclear Regulatory Commission (NRC) in Regulatory Guide (RG) 1.183 (Reference 1) and Standard Review Plan (SRP) 15.0.1 (Reference 2). NMC makes no request herein for modifications implementing the requested alternative source term (AST), we intend to submit such requests separately. As stated in Attachment 1, NMC requests NRC approval for selective implementation of AST.

The Plant Operations Review Committee (PORC) reviewed and the NMC Joint Offsite Review Committee (JOSRC) approved this proposal concluding that it is consistent with KNPP design considerations, NRC commitments, and regulations of the NRC and other governmental bodies.

This letter seeks NRC permission to make the foregoing changes and respectfully requests that the NRC grant approval for this change by December 15, 2002 as this reports approval is necessary for the approval of the Reload Transition Safety Report. The RTSR is needed for our Refueling Outage starting April 2003.

A001

Nothing in this letter should be construed to constitute a commitment or redefine a margin of safety unless specifically so stated in separate correspondence or in a safety analysis of record. This request is not risk informed.

In accordance with 10 CFR 50.30(b), the original copy of this request is signed under oath or affirmation by an officer of NMC. Additionally, NMC has transmitted a copy of this license amendment request to the State of Wisconsin as required by 10 CFR 50.91(b)(1).

If there are questions regarding this amendment, please contact either Mr. Thomas J. Webb at (920) 388-8537 or Mr. Gerald Riste at (920) 388-8424.

Sincerely,

Mark E. Warner
Site Vice President – Kewaunee and Point Beach
Nuclear Management Company, LLC

GOR

Attachments:
1. Description of Change, Safety Evaluation, Significant Hazards Determination, and Statement of Environmental Considerations
2. Westinghouse "Engineering Report for the Radiological Consequences of Accidents Using Alternative Source Term Methodology (Regulatory Guide 1.183) for the Kewaunee Nuclear Power Plant"

cc - US NRC Region III
US NRC Senior Resident Inspector
Electric Division, PSCW

Subscribed and Sworn to
Before Me This 19th Day
of March 2002

Notary Public, State of Wisconsin

My Commission Expires:
February 27, 2005

ATTACHMENT 1

Letter from M. E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

March 19, 2002

<u>Revise Kewaunee Nuclear Power Plant Accident Source Term</u>

Description of Proposed Changes

Safety Evaluation

Significant Hazards Determination

Environmental Consideration

INTRODUCTION

Nuclear Management Company, LLC (NMC) intends to revise the current Kewaunee Nuclear Power Plant (KNPP) accident source term used in KNPP design basis radiological analyses.

This request contains an evaluation of the consequences of design basis accidents previously analyzed in the KNPP safety analysis report. In accordance with provisions set forth in 10 CFR 50.67, "Accident Source Term," NMC will demonstrate with reasonable assurance that:

1. A person located at the boundary of the KNPP exclusion area during any 2-hour period following the onset of the postulated fission product release, would receive a radiation dose not exceeding 25 Roentgen-Equivalent-Man (REM) total effective dose equivalent (TEDE).
2. A person located at the outer boundary of the KNPP low population zone, who is exposed to a radioactive cloud emitted by the postulated fission product release for the entire time of its passage, would receive a radiation dose not exceeding 25 REM TEDE.
3. Adequate radiation protection is provided to permit access to and occupancy of the KNPP control room for the entire duration of the postulated accident, with no person in the control room receiving radiation exposure that exceeds 5 REM TEDE.

The current KNPP source term was developed using Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and test Reactor Sites," (Reference 1) and other early guidance. As stated in NUREG-1465 (Reference 2), "During the past 30 years substantial additional information on fission product releases has been developed based on significant severe accident research." This research "now allows more realistic estimates of the "source term" release into containment, in terms of timing, nuclide types, quantities, and chemical form, given a severe core-melt accident." As such, § 50.67 allows plants whose initial operating license was issued before January 10, 1997, to seek to revise their current accident source term used in their design basis radiological analysis.

NMC is voluntarily applying for NRC permission to replace the current source term with the alternative source term described in Attachment 2 to take advantage of these more realistic post-accident release factors. The alternate source term proposed in this request adheres to NRC guidance provided by Regulatory Guide (RG) 1.183 (Reference 3) and NUREG-0800, Standard Review Plan (SRP) 15.0.1 (Reference 4), except as otherwise stated.

NMC respectfully requests that the NRC provide its approval of this amendment by December 15, 2002, for use in conjunction with other amendments and the planned Cycle 26 Westinghouse 422V+ fuel Reload Transition Safety Report (RTSR). Unit return to service from the Cycle 26 refueling outage is planned for April 2003.

DESCRIPTION OF REQUESTED AMENDMENT

The current KNPP design basis accident source term uses methodologies and assumptions derived from TID-14844 (Reference 1). The industry has achieved significant advances in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents since publication of TID-14844. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 2). NUREG-1465 offers a representative accident source term characterized by application of a better understanding of the composition and magnitude of the radioactive material, chemical and physical properties of the material, and timing of its release to containment. Although the TID-14844 based source term remains adequate to protect public health and safety, NMC is proposing to voluntarily amend the facility design basis to use an alternative source term (AST), as provided by 10 CFR 50.67, "Accident Source Term."

The source term is a fundamental assumption used in KNPP design analyses, from which design engineers derive many aspects of facility operation. NMC is proposing an AST capable of supporting full implementation to revise the KNPP licensing bases to specify the AST in place of the original accident-source-term and establish Total-Effective-Dose-Equivalent (TEDE) dose as the new acceptance criteria. Among the re-analyses of design basis accidents in this request, NMC includes re-analysis of the Loss of Coolant Accident (LOCA) using guidance set forth in Appendix A of Regulatory Guide 1.183. Future revisions of KNPP design basis analyses will use the updated, approved, assumptions and criteria.

NUREG-1465 defines an AST model for evaluating the radiological consequence of a postulated Large Break Loss-of-Coolant Accident with core-melt. It is also the model used to determine the radiological consequence of other design basis accidents as set forth in RG 1.183. NMC used this AST methodology to evaluate KNPP design basis accidents in support of proposed changes to plant design and operation planned for future submittals, and to analyze radiological consequences of:

- Large Break Loss-of-Coolant Accident (LBLOCA),
- Steam Generator Tube Rupture (SGTR),
- Reactor Coolant Pump Locked Rotor,
- Rod Ejection,
- Fuel handling Accident (FHA),
- Main Steam-Line Break (MSLB),
- Gas Decay Tank (GDT) Rupture, and
- Volume Control Tank (VCT) Rupture.

Each of these accidents and their specific assumptions are described in detail in the Westinghouse AST Dose Analysis Report (Reference 5), attached hereto.

ACCIDENT SOURCE TERM

Reactor-core fission-product inventory available for release to containment as postulated in this request assumes maximum full power operation, as licensed. NMC calculates core power as the current licensed rated thermal power of 1650 MWt times the ECCS evaluation uncertainty[1], yielding 1683 MWt. Analyses presented herein incorporate margin including a 10% increase in source term for all events and increases in mass transfer data for most events. The assumed period of irradiation is sufficient to allow dose-significant radionuclide activity to reach equilibrium or to reach maximum values.

The attached Westinghouse report (Reference 5) provides details of the LOCA and Non-LOCA design basis accident analyses performed according to guidelines set forth in RG 1.183 and NUREG-1465.

DOSE CALCULATION

NMC dose calculations using the proposed AST apply TEDE acceptance criteria. These calculations assume that a person located at, or beyond, the exclusion area boundary (EAB) will receive a dose that is the sum of committed-effective-dose-equivalent (CEDE) from inhalation and the deep-dose-equivalent (DDE) from external exposure. They consider radionuclides, including progeny from decay of parent radionuclides that are significant to dose consequence and released radioactivity.

Analyses consider the radionuclides listed in Table 5, § 3.4, of RG 1.183, and assume that design-basis-accidents release fission-products to containment in particulate form, except for elemental iodine, organic iodine, and noble gases. Radioiodine fractions released to containment in a postulated accident are assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodine, including both gap releases and fuel pellets releases. As indicated in specific instances, transport models may affect radioiodine fractions.

Calculation of TEDE, CEDE, and DDE values for off-site-dose follows, except as otherwise stated, the guidelines of Regulatory Positions cited in RG 1.183, as does Control-Room-dose and other dose calculations. Table 1 is a summary of the Alternate Source Term analysis results. Attachment 2 explains these results in more detail.

ASSUMPTIONS AND METHODOLOGIES

Safety evaluations required by 10 CFR 50.34 form the basis for determinations made under 10 CFR 50.92 and § 50.59. Under 10 CFR 50.67, applicants for AST are required to re-analyze these safety evaluations. NMC has re-analyzed, prepared, and will maintain the 10 CFR 50.34 safety evaluations in compliance with the KNPP Quality Assurance Program. KNPP design-basis analyses presented herein use the assumptions and models selected by the NRC staff and defined in RG 1.183, except where otherwise stated, to provide appropriate and prudent safety margins.

---

[1] The basic uncertainty factor used in determining core inventory is 1.02 as provided in Appendix K to 10 CFR Part 50

Except as otherwise stated, NMC takes credit for Engineered Safety Features and other appropriately qualified, safety-related, accident mitigation features. In some cases, NMC has opted to not take credit for a qualified accident mitigation feature in order to provide an additional measure of conservatism. The Westinghouse report (Reference 5) describes these exceptions. Selected numeric input values are conservative to assure conservative postulated dose. Except as otherwise required by regulatory guidance, analyses use Technical Specification values where indicated.

Implementation of AST is a significant change to the KNPP design basis. NMC has re-analyzed the design-basis accidents included with this proposal, and the assumptions and methods used are compatible with the requested AST and TEDE criteria. Implementation of AST included in future requests will require re-analysis to employ AST and TEDE criteria in the affected design basis transients.

Meteorological data collected in accordance with the site-specific meteorological measurements program described in the KNPP USAR was used in generating accident χ/Q values. NMC applied this data using atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room, licensed by the NRC, to perform the radiological analyses identified by RG 1.183.

NMC PLANNED USES FOR ALTERNATE SOURCE TERM ANALYSIS

NMC will use the results of this analysis for implementation in a limited subset of design basis analysis. These analyses include offsite dose consequences and control room dose consequences.

SAFETY EVALUATION FOR PROPOSED ALTERNATIVE SOURCE TERM

10 CFR 50.67 states in part, "The requirements of this section apply to all holders of operating licenses issued prior to January 10, 1997... who seek to revise the current accident source term used in their design basis radiological analyses." The Nuclear Regulatory Commission (NRC) issued Kewaunee Nuclear Power Plant (KNPP) Facility Operating License DPR-43 on December 21, 1973. Nuclear Management Company, LLC, (NMC), the licensed operator of KNPP, hereby voluntarily seeks NRC approval of an alternative source term (AST). NMC will use this AST for assessment of the radiological consequence of design basis accidents (DBA) previously analyzed in the KNPP safety analysis report.

NMC used NUREG-1465 and RG 1.183 to evaluate current design-basis accidents, applying the proposed AST and TEDE dose criteria. This evaluation considered radiological dose at the site boundary, in the low population zone, and in the plant control room.

Use of an alternative source term substantially similar to that described by the NRC in NUREG-1465 and RG 1.183 does not alter actual accident sequence and progression, and cannot increase the core damage frequency (CDF), the large early release frequency (LERF), or other non-radiological impacts. Since this request does not propose modifications to existing plant design other than AST, there is no effect on CDF, LERF or other non-radiological considerations, and this request does not cause significant affect on the environment. The requested AST does not alter existing design basis accident and transient assumptions other than those included in this request, preserves effectiveness of accident mitigation systems and, hence, does not involve an unreviewed safety question.

Significant Hazards Determination for Proposed Alternative Source Term

NMC reviewed the proposed change in accordance with provisions of 10 CFR 50.92 and determined that it creates no significant hazard. The proposed change does not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Differences between the original source term and the proposed AST cannot affect the previously analyzed core damage frequency (CDF) and large early release frequency (LERF). Since there are no modifications proposed with this request for AST, Limiting Safety System Settings and Safety Limits specified in the Technical Specifications remain unchanged. Re-analysis of design basis accidents as described herein demonstrates that regulatory dose acceptance criteria continue to be satisfied. Thus, nothing in this proposal will cause an increase in the probability or consequence of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   There are no physical changes to the plant associated with this request, and the plant conditions for which NMC evaluated design-basis accidents remain valid. Consequently, this proposal introduces no new failure modes. Thus, this proposal does not create the possibility of a new or different kind of accident.

3) Involve a significant reduction in the margin of safety.

   The revised design-basis accident offsite and control-room dose-calculations proposed herein remain within regulatory acceptance criteria set forth in 10 CFR 100 and 10 CFR 50 Appendix A, General Design Criterion 19. They also use the TEDE dose acceptance criteria as directed by the Commission. An acceptable margin of safety is inherent in the limits described thereby. Thus, changes proposed by this request do not involve a significant reduction in the margin of safety.

Environmental Considerations

This amendment request proposes to modify the design basis source term according to provisions set forth in 10 CFR 50.67, "Alternative Source Term." It proposes no modification to the current plant design-basis other than AST and there is no effect on Core Damage Frequency (CDF), Large Early Release Frequency (LERF), offsite or onsite radiation doses, or other non-radiological considerations. The proposed amendment involves no significant hazard, no significant change in the types of effluents that may be released offsite, and there is no significant increase in the individual or cumulative occupational radiation exposure. Thus, this request causes no significant effect on the environment. This proposed amendment accordingly meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). In accordance with provisions of 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for this amendment.

Table 1
Kewaunee Nuclear Power Plant
Accident Source Term (AST)
Consequence Analysis in REMs

Locations	Accidents/Dose Acceptance Criteria TEDE ≤ 25 REM, (2.5* or 6.3+, for these specific accidents) Control Room ≤ 5.0 REM									
	LOCA	SGTR		*Locked Rotor	+Rod Ejection	+Fuel Handling	SLB		+Gas Decay Tank	+Volume Control Tank
		Pre Iodine Spike	*Accident Iodine Spike				Pre Iodine Spike	*Accident Iodine Spike		
Control Room	4.3	3.0	1.0	4.3	1.9	1.0	0.7	2.3	0.1	0.05
Site Boundary	0.7	1.3	0.8	1.7	0.5	0.6	0.05	0.2	0.1	0.1
Low Population Zone	0.15	0.3	0.2	0.3	0.11	0.11	0.02	0.05	0.02	0.01

References

1. USAEC Technical Information Document TID-14844, "Calculation of Distance Factors for Power and test Reactor Sites," by J. J. DiNunno, et al, for U. S. Atomic Energy Commission, dated 1962

2. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," by L. Soffer, S. B. Burson, C. M. Ferrell, R. Y. Lee, J. N. Ridgely, dated February 1995

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

4. NUREG 0800, Standard Review Plan, SRP-15.0.1, Rev. 0 "Radiological Consequence Analyses Using Alternative Source Terms," dated July 2000

5. Westinghouse letter, KEW-LTR-ESI-02-055, KEW-UPRATE-02-008, "Kewaunee Nuclear Power Plant 7.4% Power Uprate Project, Alternate Source Term (R.G. 1.183) Dose Analysis Report," dated March 18, 2002, which transmitted Westinghouse report, "Engineering Report for the Radiological Consequences of Accidents Using Alternative Source Term Methodology (Regulatory Guide 1.183) for the Kewaunee Nuclear Power Plant," dated March 15, 2002.

ATTACHMENT 2

Letter from M. E. Warner (NMC)

To

Document Control Desk (NRC)

Dated

March 19, 2002

Revise Kewaunee Nuclear Power Plant Accident Source Term

Westinghouse "Engineering Report for the Radiological Consequences of Accidents Using Alternative Source Term Methodology (Regulatory Guide 1.183) for the Kewaunee Nuclear Power Plant"

Engineering Report for the Radiological Consequences of Accidents Using Alternative Source Term Methodology (Regulatory Guide 1.183) for the Kewaunee Nuclear Power Plant

March 15, 2002

Prepared by: U. Bachrach
Containment and Radiological Analysis

Reviewed by: J. L. Grover
Containment and Radiological Analysis

Table of Contents

1.0 Radiological Consequences Utilizing Alternative Source Terms

1.1 Introduction

The Kewaunee Nuclear Power Plant (KNPP) licensing basis for the radiological consequences analyses for Chapter 14 of the UFSAR 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). This includes modeling consistent with NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 4). The alternative source term methodology as established in RG 1.183 (based on NUREG-1465) is being used to calculate the offsite and control room radiological consequences for the KNPP to support the control room habitability program. The following UFSAR chapter 14 radiological consequences analyses are analyzed: LBLOCA, Steam Generator Tube Rupture (SGTR), Locked Rotor, Rod Ejection, Fuel Handling Accident (FHA), Main Steamline Break (MSLB), Gas Decay Tank (GDT) Rupture, and Volume Control Tank (VCT) Rupture. Each accident and the specific input assumptions are described in detail in subsequent sections in this report.

1.2 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 through 8.

The total effective dose equivalent (TEDE) doses are determined at the site boundary (SB) for the limiting two hour interval, at the low population zone (LPZ) and to control room personnel (CR) for the duration of the event. The dose conversion factors (DCFs) used in determining the committed effective dose equivalent (CEDE) or inhalation dose are from Reference 5. The TEDE dose is equivalent to the CEDE dose plus the acute dose for the duration of exposure to the cloud. The CEDE dose DCFs are given in Table 1.

The γ-body (acute) doses are based on the average disintegration energies from Reference 6 for the iodine isotopes and from Reference 7 for the remainder of the nuclides (except the noble gases). The dose conversion factors for the noble gases are taken from ICRP Publication 30 (Reference 8). The average disintegration energies and dose conversion factors for the γ-body doses are listed in Table 2.

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

Parameters modeled in the control room personnel dose calculations are provided in Table 4. These parameters include the normal operation flowrates, the emergency operation flowrates, control room volume, filter efficiencies and control room operator breathing rates. In the analyses presented in this report, the control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for release of activity from the release point to the control room intake are used to determine the activity available at the intake. 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 5 for all nuclides. 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 I-131 for iodine nuclides is provided in Table 6. The core and coolant activities in Tables 5 and 6 are based on a core power of 1650 MWt increased to 1683 to cover 2% uncertainty. These activities are conservatively increased by an additional 10% prior to input into the analyses presented in this report to allow this analysis to cover future uprating of the plant. Decay constants for each nuclide are provided in Table 7.

1.3 Dose Calculation Models

1.3.1 Offsite Dose Calculation Models

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 1)

$(IAR)_{ij}$ = integrated activity of isotope i released during the time interval j (Ci)

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 3)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table 3)

Offsite gamma doses from noble gases are calculated using the following equation:

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

where:

$D\gamma$ = gamma dose via cloud submersion (rem)

DCF_i = gamma dose conversion factor via external exposure for isotope i ($rem \cdot m^3/Ci \cdot sec$) (Table 2)

$(IAR)_{ij}$ = 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 3)

Offsite gamma doses from other isotopes are calculated using the following equation:

$$D_\gamma = 0.25 \sum_i \left[\bar{E}\gamma_i \left(\sum_j (IAR)_{ij} (\chi/Q)_j \right) \right]$$

where:

$D\gamma$ = gamma dose via cloud submersion (rem)

$\bar{E}\gamma_i$ = gamma disintegration energy for isotope i (mev/dis) (Table 2)

$(IAR)_{ij}$ = 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 3)

1.3.2 Control Room Dose Calculation Models

CEDE doses (due to inhalation) and gamma doses (due to external exposure) are calculated for 30 days in the control room.

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 filtered recirculation flow are used to calculate the concentration of activity in the control room. Control room parameters used in the analyses are presented in Table 4.

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 4)

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

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table 4)

Control room external exposure doses due to noble gas activity in the control room volume are calculated using the following equation:

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

where:

$D\gamma$ = gamma dose via cloud submersion (rem)

GF = geometry factor, calculated based on Reference 11, using the equation: $GF = \frac{1173}{V^{0.338}}$, where V is the control room volume in ft^3

DCF_i = gamma dose conversion factor via external exposure for isotope i (rem·m^3/Ci·sec) (Table 2)

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

Control room external exposure doses due to activity in the control room volume, other than noble gas activity, are calculated using the following equation:

$$D_\gamma = 0.25\left(\frac{1}{GF}\right) * \sum_i \bar{E}\gamma_i \left(\sum_j \text{Conc}_{ij}\right)$$

where:

$D\gamma$ = gamma dose via cloud submersion (rem)

GF = geometry factor, calculated based on Reference 11, using the equation:

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

$\bar{E}\gamma_i$ = gamma disintegration energy for isotope i (mev/dis) (Table 2)

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

2.0 Large Break Loss of Coolant Accident Radiological Analysis

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.

2.1 Comparison of RG 1.183 Source Term Methodology to TID-14844

The reanalysis of the LBLOCA offsite and control room doses for KNPP uses the following RG 1.183 source term characteristics in place of those identified in TID-14844 and Regulatory Guide 1.4 (Reference 12):

Iodine chemical species
Fission product release timing
Fission product release phases through early in-vessel
Fission product release fractions
Fission product groups

A comparison of RG 1.183 to TID-14844 is provided in Tables 8 through 11.

2.2 Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 12. Activity from the reactor coolant system and melted fuel is released into the containment. The analysis considers the release of activity from the containment via containment leakage. In addition, once recirculation 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 outside containment in the Auxiliary Building. 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.

2.2.1 Source Term

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 UFSAR.

- 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 11 lists the nuclides being considered for the LOCA with core melt (eight groups of nuclides). Tables 9 and 10 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 and the fraction that is in the organic form is much smaller. The iodine characterization from RG 1.183 is compared in Table 8 with that from Regulatory Guide 1.4.

  The other groups of nuclides (other than the iodines and the noble gases) all occur as particulates only.

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 analysis, 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.

2.2.2 Containment Modeling

The Containment System of the Kewaunee plant consists of three buildings: the Reactor Containment Vessel, the Shield Building and the Auxiliary Building. The containment and shield buildings are modeled as discrete volumes, which consider hold-up, removal and decay. Since no hold-up is modeled for the auxiliary building, a separate volume representing the auxiliary building is not included in the model.

The containment is assumed to leak at the design leak rate of 0.5% per day for the first 24 hours of the accident and then to leak at half that rate (0.25% per day) for the remainder of the 30 day period following the accident considered in the analysis.

During the first 10 minutes of the accident, it is assumed that 90 percent of the activity leaking from the containment is discharged directly to the environment and 10 percent enters the Auxiliary Building where it is released through filters. After 10 minutes, only 1.0 percent of the activity leaking from the containment is assumed to go directly to the environment, 10 percent continues to go to the Auxiliary Building, and 89 percent is assumed to pass into the Shield Building. The air discharged from the Shield Building is filtered to remove iodine. Additionally, once the Shield Building is brought to sub-atmospheric pressure at 30 minutes into the event, the iodine is subject to removal by recirculation through filters. A shield building participation fraction of 0.5 is assumed.

2.2.3 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 with no startup delay. Earlier spray actuation is conservative since it results in earlier spray termination. There is no benefit from the earlier actuation since there is little activity in the containment at the time the sprays start. When the

RWST drains to a predetermined setpoint level, the operators switch to recirculation of sump liquid to provide a source for the sprays. Switchover to recirculation spray is not credited in the analysis and all spray is assumed to be terminated when the RWST drains down. The analysis conservatively assumed that the sprays are terminated 0.91 hours from the start of the event.

2.2.3.1 Containment Spray Removal of Elemental Iodine

The current Standard Review Plan (Reference 9) 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

Parameters for KNPP are listed below:
K_g = 9.84 ft/min
T = 13 sec
F = 1300 gpm
V = 1.32×10^6 ft^3
D = 0.124 cm

These parameters and the appropriate conversion factors were used to calculate the elemental spray removal coefficients. The upper limit of 20 hr^{-1} specified for this model is applied in the analysis in place of the calculated value of > 24 hr^{-1}.

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 does not occur before spray termination at 0.91 hours.

2.2.3.2 Containment Spray Removal of Particulates

Particulate spray removal is determined using the model described in Reference 9. The first order spray removal rate constant for particulates may be written as follows:

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

Where: h = Drop Fall Height
F = Spray Flow Rate
V = Volume Sprayed

E = Single Drop Collection Efficiency
d = Drop Diameter

Parameters for KNPP are listed below:
H = 65 ft
F = 1300 gpm
V = 1.32×10^6 ft^3

The E/d term depends upon the particle size distribution and spray drop size. From Reference 10 it is conservative to use 10 m^{-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.

These parameters and the appropriate conversion factors were used to calculate the particulate spray removal coefficients. A conservative value of 5.0 hr^{-1} was used in the analysis. The airborne inventory does not drop to 2 percent of the total particulate iodine released to the containment (this is a DF of 50) before spray termination at 0.91 hours.

2.2.3.3 Sedimentation Removal of Particulates

During spray operation, no credit is taken for sedimentation removal of particulates, although it would take place. It is assumed that containment spray operation is terminated at 0.91 hours. Recirculation sprays are not credited. Credit is taken for sedimentation removal of particulates after spray termination.

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 10). The CSE tests determined that there was a significant sedimentation removal rate even with a relatively low aerosol concentration. From Reference 10, even at an air concentration of 10 μg/m^3, the sedimentation removal coefficient was above 0.3 h^{-1}. With 2.0 percent of particulates remaining airborne at the end of credited spray removal, there would be more than 10,000 μg/m^3 and an even higher sedimentation rate would be expected. (This conservatively assumes that the DF of 50 is reached, which is not the case as noted above.) For the analysis the sedimentation removal coefficient is conservatively assumed to be only 0.1 h^{-1}. This value for sedimentation removal of particulates has been accepted by the NRC for Indian Point Unit 2 and Shearon Harris in their Safety Evaluation Reports for the application of the alternate source term methodology. It is also conservatively assumed that sedimentation removal does not continue beyond a DF of 1000.

2.2.4 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). Although recirculation is not initiated until the RWST has drained to the pre-determined setpoint level (at about 0.91 hours) the analysis conservatively considers leakage from the start of the event.

The leakage to the auxiliary building is 6 gallon/hr. In the licensing basis analysis of record a 10% of the iodine contained in the leak flow becomes airborne. For this new analysis the airborne fraction is conservatively increased to 10% when the sump temperature is above 212°F. Once the sump solution temperature drops below 212°F, the airborne fraction is reduced to analysis of record value of 1%. The reduction in airborne fraction is conservatively delayed until 1.5 hours from the start of the event.

RHR back-leakage to the RWST is assumed at a rate of 3 gpm for the first 24 hours and 1.5 gpm for the remainder of the event. It is assumed that 1% of the iodine contained in the leak flow becomes airborne. The 1% value is applied even when the sump is above 212°F since any incoming water would be cooled by the water remaining in the RWST. The RWST vents to the auxiliary building.

It is assumed that half the iodine activity that becomes airborne in the auxiliary building from the two leak sources is removed by plateout on surfaces. Releases from the auxiliary building are subject to filtration by the auxiliary building special ventilation system.

For the analysis, all iodine activity is assumed in elemental form. In the RG 1.183 source term methodology, the iodine is mainly in the form of particulate cesium iodide which would not become airborne. However, to bound a condition where the recirculation liquid was in a low pH environment and changed form, all of the leakage is assumed to be elemental.

2.2.5 Control Room Isolation

In the event of a Large Break LOCA, the low pressurizer 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 accident mode of operation. It is conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 2 minutes after event initiation.

2.3 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.

2.4 Results and Conclusions

The large break LOCA doses are:

Site Boundary	0.7 rem TEDE
Low Population Zone	0.15 rem TEDE
Control Room	4.3 rem TEDE

The acceptance criteria are met.

The Site Boundary dose reported is for the worst two hour period, determined to be from 1.8 hours to 3.8 hours.

3.0 Steam Generator Tube Rupture Radiological Analysis

The steam generator tube rupture (SGTR) event is separated into two analyses, a thermal and hydraulic analysis and a radiological consequences analysis. The thermal hydraulic analysis is not impacted by the alternative source term methodology, so the results from the licensing basis analysis, summarized below, are used as input to the dose analysis.

3.1 Steam Generator Tube Rupture Thermal and Hydraulic Analysis Results

The major hazard associated with an SGTR event is the radiological consequences resulting from the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator (SG) and subsequent release of radioactivity to the atmosphere. The primary thermal-hydraulic parameters which affect the calculation of offsite doses for an SGTR include the amount of reactor coolant transferred to the secondary side of the ruptured steam generator, the amount of primary to secondary break flow that flashes to steam and the amount of steam released from the ruptured steam generator to the atmosphere.

Based on a primary and secondary side mass and energy balance, the break flow and atmospheric steam releases from the ruptured and intact steam generators were calculated for 30 minutes. After 30 minutes, it was assumed that steam is released only from the intact steam generator in order to dissipate the core decay heat and to subsequently cool the plant down to the RHR system operating conditions. For KNPP, it was assumed that plant cooldown to RHR operating conditions can be accomplished within 8 hours after initiation of the SGTR event and that steam release is terminated at this time. A primary and secondary side mass and energy balance was used to calculate the steam release from the intact steam generator from 0 to 2 hours and from 2 to 8 hours.

The limiting tube rupture break flow, break flow flashing fraction and ruptured steam generator atmospheric steam releases from 0 to 30 minutes are provided in Table 13 along with the long term intact steam generator steam releases for use in radiological consequences analysis. The values in Table 13 include an approximate 10% increase in mass flow rates for use in the conservative radiological analysis. Increasing the mass transfer data before performing the radiological consequences analysis allows future plant changes that result in small increases in the mass transfer rates to be evaluated without requiring the radiological analysis to be redone.

3.2 Steam Generator Tube Rupture Radiological Analysis

For the SGTR, the complete severance of a single steam generator tube is 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 the main condenser, the atmospheric dump valves, or the safety valves (MSSVs). In addition, iodine 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 from the SGs following the accident.

3.2.1 Input Parameters and Assumptions

A summary of input parameters and assumptions is provided in Table 14.

The analysis of the SGTR radiological consequences uses the analytical methods and assumptions outlined in RG 1.183. For the pre-accident iodine spike case it is assumed that a reactor transient has occurred prior to the SGTR 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 SGTR creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 335 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 μCi/gm of DE I-131. For this analysis, the spike factor was conservatively increased from 335 to 500. The duration of the accident-initiated iodine spike is limited by the amount of activity available in the fuel-clad gap. Based on having 10 percent of the iodine in the fuel-clad gap, the gap inventory would be depleted within 4.0 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 iodine activity concentration of the secondary coolant at the time the SGTR 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 in the intact SG is assumed to be 500 gpd. This bounds the Technical Specification limit of 150 gpd.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 is taken for steam released to the condenser.

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

Break flow flashing fractions and steam release rates from the intact and ruptured steam generator were calculated. The amount of break flow that flashes to steam is conservatively calculated assuming that all break flow is from the hot leg side of the break and that the primary temperatures remain constant.

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.

3.2.2 Control Room Isolation

The low pressurizer pressure SI setpoint will be reached at ~2.9 minutes from event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. It is conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 5 minutes after event initiation.

3.3 Acceptance Criteria

The offsite dose limit for a SGTR with a pre-accident iodine spike is 25 rem TEDE per RG 1.183. This is the guideline value of 10CFR50.67. For a SGTR 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.

3.4 Results and Conclusions

The SGTR accident doses are listed below.

For the pre-accident iodine spike:

Site Boundary	1.3 rem TEDE
Low Population Zone	0.3 rem TEDE
Control Room	3.0 rem TEDE

For the accident-initiated iodine spike:

Site Boundary	0.8 rem TEDE
Low Population Zone	0.2 rem TEDE
Control Room	1.0 rem TEDE

The acceptance criteria are met.

The Site Boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

4.0 Locked Rotor Radiological Analysis

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 atmosphere as a result of steaming from the steam generators following the accident.

4.1 Input Parameters and Assumptions

A summary of input parameters and assumptions is provided in Table 15.

The analysis of the locked rotor radiological consequences uses the analytical methods and assumptions outlined in RG 1.183.

It is conservatively assumed that 100% 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. Eight percent of the total core activity of iodine, five 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.

It is assumed that a reactor transient has occurred prior to the locked rotor 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. 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 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 500 gpd. This bounds the Technical Specification limit of 150 gpd/SG.

An iodine partition factor in the SGs of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. This partition factor is applied to alkali metals. Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 could be taken for steam released to the condenser, but this is conservatively ignored.

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 KNPP 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. The calculated values were increased by 10% for input to the radiological analysis. Increasing the mass transfer data before performing the radiological consequences analysis allows future plant changes that result in small increases in the mass transfer rates to be evaluated without requiring the radiological analysis to be redone.

4.1.1 Control Room Isolation

It is assumed that the control room HVAC system begins in normal operation mode. The activity level in the air supply duct causes a high radiation signal almost immediately. It is conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 1 minute after event initiation.

4.2 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.

4.3 Results and Conclusions

The locked rotor doses are:

Site Boundary	1.7 rem TEDE
Low Population Zone	0.3 rem TEDE
Control Room	4.3 rem TEDE

The acceptance criteria are met.

The Site Boundary doses reported are for the worst two hour period, determined to be from 6.0 to 8.0 hours.

5.0 Rod Ejection Radiological Analysis

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 main condenser or the atmospheric relief valves for 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.

5.1 Input Parameters and Assumptions

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 16.

5.1.1 Source Term

Less than 15% of the fuel rods in the core undergo DNB as a result of the rod ejection accident. In determining the offsite doses following rod ejection accident, it is conservatively assumed that 15% of the fuel rods in the core suffer sufficient damage that all of their gap activity is released. 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 1.7 was applied.

A small fraction of the fuel in the failed fuel rods is assumed to melt as a result of the rod ejection accident. It is assumed that 0.375% of the core melts. This is based on the assumption that 50% of the rods in DNB undergo centerline melting, with the melting limited to the inner 10% and occurring over 50% of the axial length of the affected rods.

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

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

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

It is assumed that a reactor transient has occurred prior to the rod ejection rotor 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. 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.

5.1.2 Containment Release Pathway

The Containment System of the Kewaunee plant consists of three buildings: the Reactor Containment Vessel, the Shield Building and the Auxiliary Building. The containment and shield buildings are modeled as discrete volumes, which consider hold-up, removal and decay. Since no removal or hold-up is modeled for the auxiliary building, a separate volume representing the auxiliary building is not included in the model.

The containment is assumed to leak at the design leak rate of 0.5% per day for the first 24 hours of the accident and then to leak at half that rate (0.25% per day) for the remainder of the 30 day period following the accident considered in the analysis.

During the first 10 minutes of the accident, it is assumed that 90 percent of the activity leaking from the containment is discharged directly to the environment and 10 percent enters the Auxiliary Building where it is released through filters. After 10 minutes, only 1.0 percent of the activity leaking from the containment is assumed to go directly to the environment, 10 percent continues to go to the Auxiliary Building, and 89 percent is assumed to pass into the Shield Building. The air discharged from the Shield Building is filtered to remove iodine. Additionally, once the Shield Building is brought to sub-atmospheric pressure at 30 minutes into the event, the iodine is subject to removal by recirculation through filters. A shield building participation fraction of 0.5 is assumed.

For the containment leakage pathway, no credit is taken for sedimentation or plateout onto containment surfaces or for containment spray operation which would remove airborne particulates and elemental iodine.

5.1.3 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 bounding time of 30 minutes was selected for this analysis, although analyses of the small break LOCA pressure transient have shown that this would occur well before that time. A 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 500 gpd. This bounds the Technical Specification limit for of 150 gpd/SG. Although the primary to secondary pressure differential drops throughout the event, the constant flow rate is 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 applied to alkali metals. Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 could be taken for steam released to the condenser, but this is conservatively ignored.

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

5.1.4 Control Room Isolation

In the small break LOCA analysis, the low pressurizer pressure SI setpoint will be reached within 40 seconds from event initiation. The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. It is conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 2 minutes after event initiation.

5.2 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.

5.3 Results and Conclusions

The rod ejection doses are:

Site Boundary	0.50 rem TEDE
Low Population Zone	0.11 rem TEDE
Control Room	1.90 rem TEDE

The acceptance criteria are met.

The Site Boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

6.0 Fuel Handling Accident Radiological Analysis

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the auxiliary building. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the fuel pool ventilation system.

6.1 Input Parameters and Assumptions

The major assumptions and parameters used in the analysis are itemized in Table 17. All activity released from the fuel pool is assumed to be released to the atmosphere in two hours, using an exponential release model with higher releases in the initial periods since this is conservative for the control room doses. No credit is taken for the spent fuel pool ventilation system operation in the Auxiliary Building. No credit is taken for isolation of containment for the FHA in containment. Since the assumptions and parameters for a FHA inside containment are identical to those for a FHA in the Auxiliary Building, the radiological consequences are the same regardless of the location of the accident.

6.1.1 Source Term

The calculation of the radiological consequences following a fuel handling accident (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 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 1.7 times the core average power.

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

6.1.2 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.

6.1.3 Pool Scrubbing Removal of Activity

Per the technical specifications, it is assumed that there is a minimum of 23 feet of water above the fuel. With this water depth, the overall pool decontamination factor (DF) for iodine is 200. The DF for organic iodine and noble gases is 1.0. [The elemental iodine scrubbing factor is conservatively set to provide the overall effective DF of 200 specified in RG 1.183. By back calculating, it is approximately 286. This is well below the value of 500 indicated in the RG and conservatively accounts for increased rod internal pressures above 1200 psig.]

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

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

6.1.4 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.

6.1.5 Control Room Isolation

It is assumed that the control room HVAC system begins in normal operation mode. The activity level in the air supply duct causes a high radiation signal almost immediately. It is conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 1 minute after event initiation.

6.2 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.

6.3 Results and Conclusions

The fuel handling accident doses are:

Site Boundary	0.6 rem TEDE
Low Population Zone	0.11 rem TEDE
Control Room	1.0 rem TEDE

The acceptance criteria are met.

The Site Boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

7.0 Steam Line Break Radiological Analysis

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 into 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.

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

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 10 percent of the iodine in the fuel-clad gap, the gap inventory would be depleted within 4.0 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 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 500 gpd. This bounds the Technical Specification limit for of 150 gpd/SG.

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.

An iodine partition factor in the intact SG of 0.01 (curies iodine/gm steam) / (curies iodine/gm water) is used. Prior to reactor trip and concurrent loss of offsite power an iodine removal factor of 0.01 could be taken for steam released to the condenser, but this is conservatively ignored.

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 generator. The calculated intact steam generator steam release values were increased by 10% for input to the radiological analysis. Increasing the mass transfer data before performing the radiological consequences analysis allows future plant changes that result in small increases in the mass transfer rates to be evaluated without requiring the radiological analysis to be redone.

Within 72 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.

7.1.1 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 accident mode of operation. It is conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 5 minutes after event initiation.

7.2 Acceptance Criteria

The offsite dose limit for a SLB with a pre-accident iodine spike 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.

7.3 Results and Conclusions

The SLB accident doses are listed below.

For the pre-accident iodine spike:

Site Boundary	0.05 rem TEDE
Low Population Zone	0.02 rem TEDE
Control Room	0.7 rem TEDE

For the accident-initiated iodine spike:

Site Boundary	0.2 rem TEDE
Low Population Zone	0.05 rem TEDE
Control Room	2.3 rem TEDE

The acceptance criteria are met.

The Site 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 from 4.0 to 6.0 hours for the accident initiated iodine spike.

8.0 Gas Decay Tank Rupture and Volume Control Tank Rupture Radiological Analyses

For the gas decay tank rupture, a failure is assumed that results in the release of the contents of one gas decay tank. For the volume control tank rupture, a failure is assumed that results in the release of the contents of the tank plus the noble gases and a fraction of the iodines from the letdown flow until the letdown path is isolated.

8.1 Input Parameters and Assumptions

The major assumptions and parameters used to determine the doses due to the gas decay tank failure and the volume control tank failure are given in Table 19.

The control room HVAC is modeled as being initially in the normal operating mode. Actuation of the accident mode of operation would be initiated by high radiation detected in the air supply duct. The HVAC would be in the accident mode within 30 seconds of accident initiation.

8.1.1 Gas Decay Tank Rupture

The inventory of gases in the tank are based on operation of the plant with 1.0 percent fuel defects and with no purge of activity from the volume control tank to the gas decay tank during the cycle. This is followed by shutdown of the plant and degassing of the primary coolant to the tank. A failure in the gaseous waste processing system is assumed to result in release of the tank inventory with a release duration of five minutes.

8.1.2 Volume Control Tank Rupture

The inventory of gases in the tank is based on continuous operation with 1.0 percent fuel defects and without any purge of the gas space. The inventory of iodine in the tank is based on operation of the plant with 1.0 percent fuel defects and with 90 percent of the iodine removed by the letdown demineralizer.

As a result of the accident, all of the noble gas in the tank and 1.0 percent of the iodine in the tank liquid is assumed to be released to the atmosphere over a period of 5 minutes. After event initiation, letdown flow to the volume control tank continues at the maximum flow rate of 88 gpm (maximum letdown flow plus 10-percent uncertainty) for 5 minutes when the letdown line is assumed to be isolated. The primary coolant noble gas activities used in the volume control tank rupture dose calculations are based on operation with 1.0 percent fuel defects. The primary coolant iodine activity is conservatively assumed to be at the pre-existing iodine spike level of 60 μCi/gram dose equivalent I-131, which is reduced by 90 percent by the letdown demineralizer. All of the noble gas and 1.0 percent of the iodine in the letdown flow are assumed to be released to the environment.

8.1.3 Control Room Isolation

It is assumed that the control room HVAC system begins in normal operation mode. The activity level in the air supply duct causes a high radiation signal almost immediately, for both postulated tank ruptures. It is conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 30 seconds after event initiation.

8.2 Acceptance Criteria

The offsite dose limits for a gas decay tank rupture or a volume control tank rupture are assumed to be the same as those for the fuel handling accident. That 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.

8.3 Results and Conclusions

The gas decay tank rupture doses are:

Site Boundary	0.1 rem TEDE
Low Population Zone	0.02 rem TEDE
Control Room	0.1 rem TEDE

The acceptance criteria are met.

The volume control tank rupture doses are:

Site Boundary	0.1 rem TEDE
Low Population Zone	0.01 rem TEDE
Control Room	0.05 rem TEDE

The acceptance criteria are met.

The Site Boundary doses reported are for the worst two hour period, determined to be from 0.0 to 2.0 hours.

9.0 Conclusions

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.

Implementation of this alternative source term methodology on the Kewaunee Nuclear Power Plant's design basis accidents 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, rod ejection, fuel handling accident, steamline break, and gas decay tank and volume control tank ruptures 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 Kewaunee Nuclear Power Plant's design and operation:

- Movement of fuel in the containment with the equipment hatch and/or personnel air lock open.
- Removal of the requirement for recirculation sprays in containment following a large break LOCA, for radiological concerns.
- Relaxation of the time requirements for establishing recirculation in the shield building following a large break LOCA or rod ejection accident.
- Relaxation of the Technical Specification limits for primary coolant iodine activity.
- Removal of limitation on calculated rods-in-DNB for a locked rotor.

10.0 References

1. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. AEC, Division of Licensing and Regulation, J. J. DiNunno, et. al, March 23, 1962.
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. SAE-CRA-98-104, "Revision 1 to the TITAN5 User's Guide," 4/9/98.
7. International Commission on Radiological Protection, "Radionuclide Transformations, Energy and Intensity of Emissions," ICRP Publication 38, 1983.
8. International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, 1979.
9. NUREG-0800, Standard Review Plan 6.5.2, "Containment Spray as a Fission Product Cleanup System," Revision 2, December 1988.
10. Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983
11. 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.
12. Regulatory Guide 1.4, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident," (Rev. 2, June 1974).

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

Table 2: Average Gamma Disintegration Energies and Noble Gas Dose Conversion Factors

Isotope	Energy (mev/dis)	Isotope	Energy (mev/dis)
I-131	0.38	Ru-103	0.468
I-132	2.2	Ru-105	0.775
I-133	0.6	Ru-106	0.0
I-134	2.6	Rh-105	0.078
I-135	1.4	Mo-99	0.15
		Tc-99m	0.126
Te-127	4.86E-3		
Te-127m	0.0112	Y-90	1.7E-6
Te-129m	0.0375	Y-91	3.61E-3
Te-129	0.0591	Y-92	0.251
Te-131m	1.42	Y-93	0.0889
Te-132	0.233	Nb-95	0.766
Sb-127	0.688	Zr-95	0.739
Sb-129	1.44	Zr-97	0.179
		La-140	2.31
Ce-141	0.076	La-141	0.0427
Ce-143	0.282	La-142	2.68
Ce-144	0.021	Nd-147	0.14
Pu-238	1.81E-3	Pr-143	8.9E-9
Pu-239	8.08E-4	Am-241	0.0324
Pu-240	1.73E-3	Cm-242	1.83E-3
Pu-241	2.54E-6	Cm-244	1.7E-3
Np-239	0.172		
		Sr-89	8.45E-5
Cs-134	1.55	Sr-90	0.0
Cs-136	2.16	Sr-91	0.693
Cs-137	0.564	Sr-92	1.34
Rb-86	0.0945	Ba-139	0.043
		Ba-140	0.182

Isotope	DCF (rem· m^3 /Ci · sec)	Isotope	DCF (rem· m^3 /Ci · sec)
Kr-85m	0.0307	Xe-133m	0.00553
Kr-85	0.000484	Xe-133	0.00624
Kr-87	0.146	Xe-135m	0.0775
Kr-88	0.37	Xe-135	0.0482
Xe-131m	0.00152	Xe-138	0.198

Table 3: Offsite Breathing Rates and Atmospheric Dispersion Factors

	Offsite Breathing Rates (m^3/sec)
0 - 8 hours	3.47E-4
8 - 24 hours	1.75E-4
>24 hours	2.32E-4

	Offsite Atmospheric Dispersion Factors (sec/m^3)
Site Boundary*	2.232E-4
Low Population Zone	
0 - 2 hours	3.977E-5
2 - 24 hours	4.100E-6
1 - 2 days	2.427E-6
> 2 days	4.473E-7

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

Table 4: Control Room Parameters

Volume (ft^3)	127,600
Normal Ventilation Flow Rates (cfm)	
Filtered Makeup Flow Rate	0.0
Filtered Recirculation Flow Rate	0.0
Unfiltered Makeup Flow Rate	2500 (±10%)
Unfiltered Recirculation Flow Rate	(Not modeled - no impact on analyses)
Emergency Ventilation System Flow Rates (cfm)	
Filtered Makeup Air Flow Rate	0.0
Filtered Recirculation Flow Rate	2500 (±10%)
Unfiltered Inleakage	200
Unfiltered Recirculation Flow Rate	(Not modeled - no impact on analyses)
Filter Efficiencies (%)	
Elemental	90
Organic	90
Particulate	99
R-23 Sensitivity (cpm/µCi/cc for Xe-133)	2.32E7
Setpoint for Control room isolation (cpm)	1.0E4
Location of R-23	R-23 is located at a junction of the intake and the recirculation ducts such that it monitors the mixed air stream.
Delay to Initiate Switchover of HVAC after S-signal	63 seconds
Delay to Complete Switchover of HVAC	< 10 seconds
Breathing Rate - Duration of the Event (m^3/sec)	3.47E-4
Atmospheric Dispersion Factors (sec/m^3)	
0 – 8 hours	2.93E-3
8 – 24 hours	1.73E-3
1 – 4 days	6.74E-4
4 – 30 days	1.93E-4
Occupancy Factors*	
0 - 24 hours	1.0
1 - 4 days	0.6
4 - 30 days	0.4

* These occupancy factors (from Reference 11) 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 5: Core Total Fission Product Activities

Based on 102% of 1650 MWt

Isotope	Activity (Ci)	Isotope	Activity (Ci)
I-131	4.48E7	Cs-134	9.54E6
I-132	6.49E7	Cs-136	2.49E6
I-133	9.22E7	Cs-137	6.26E6
I-134	1.02E8	Rb-86	9.66E4
I-135	8.62E7		
		Ru-103	6.73E7
Kr-85m	1.24E7	Ru-105	4.48E7
Kr-85	5.84E5	Ru-106	2.36E7
Kr-87	2.38E7	Rh-105	4.19E7
Kr-88	3.36E7	Mo-99	8.26E7
Xe-131m	5.00E5	Tc-99m	7.23E7
Xe-133m	2.87E6		
Xe-133	9.03E7	Y-90	4.86E6
Xe-135m	1.79E7	Y-91	5.82E7
Xe-135	2.51E7	Y-92	6.08E7
Xe-138	7.68E7	Y-93	6.98E7
		Nb-95	7.77E7
Te-127	4.60E6	Zr-95	7.72E7
Te-127m	6.07E5	Zr-97	7.63E7
Te-129m	2.07E6	La-140	8.34E7
Te-129	1.39E7	La-141	7.56E7
Te-131m	6.42E6	La-142	7.33E7
Te-132	6.39E7	Nd-147	3.03E7
Sb-127	4.65E6	Pr-143	6.89E7
Sb-129	1.41E7	Am-241	8.03E3
		Cm-242	2.20E6
Ce-141	7.61E7	Cm-244	1.58E5
Ce-143	7.07E7		
Ce-144	5.99E7	Sr-89	4.54E7
Pu-238	1.67E5	Sr-90	4.68E6
Pu-239	1.65E4	Sr-91	5.63E7
Pu-240	2.38E4	Sr-92	6.05E7
Pu-241	6.37E6	Ba-139	8.29E7
Np-239	8.22E8	Ba-140	8.00E7

Table 6: RCS Coolant Concentrations

Based on 1.0 μCi/gm DE I-131 for Iodines and 1% Fuel Defects for Noble Gases and Alkali Metals

Isotope	Activity (μCi/gm)
I-131	0.782
I-132	0.717
I-133	1.153
I-134	0.161
I-135	0.640
Kr-85m	1.64
Kr-85	7.51
Kr-87	1.07
Kr-88	3.09
Xe-131m	2.52
Xe-133m	3.96
Xe-133	215.0
Xe-135m	0.469
Xe-135	7.82
Xe-138	0.607
Cs-134	2.26
Cs-136	2.59
Cs-137	1.91
Rb-86	0.0232

Iodine concentrations are converted to dose equivalent (DE) I-131 using the dose conversion factors in ICRP-30 (Reference 8) for direct thyroid doses.

Table 7: 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		
		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 8: Iodine Chemical Species

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

Table 9: Fission Product Release Timing

Release Phase	Duration (TID-14844)	Duration (RG 1.183)[1]
Gap Activity	instantaneous release	0.5 hour
Early In-vessel	instantaneous release	1.3 hour

1. Releases are sequential.

Table 10: Core Fission Product Release Fractions

	Gap Release [1]		Early In-Vessel	
	TID-14844	RG 1.183	TID-14844	RG 1.183
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 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 35% in-vessel release for a total release of 40% of core.

(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 11: 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 12: Assumptions Used for Large Break LOCA Dose Analysis

Source Term	
Core Activity	See Table 5
Activity release fractions and timing	See Tables 9 & 10
Iodine chemical form in containment (%)	
Elemental	4.85
Organic	0.15
Particulate (cesium iodide)	95
Iodine chemical form in sump & recirculating liquid (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
Containment	
Containment net free volume (ft^3)	1.32E6
Shield building volume (ft^3)	3.74E5
Shield building participation fraction	0.5
Auxiliary building volume	Not Modeled, No holdup Credited
Containment leak rates (weight %/day)	
0 – 24 hours	0.5
> 24 hours	0.25
Containment Leak Path Fractions	
0-10 minutes	
Through Shield Building	0.0
Through Auxiliary Building SV	0.1
Direct to Environment	0.9
10 minutes - 30 days	
Through Shield Building	0.89
Through Auxiliary Building SV	0.10
Direct to Environment	0.01
Shield Building Air Flows (cfm)	
0-10 minutes	
Shield Building to Environment	Not applicable
Shield Building Recirculation	Not applicable
10 minutes – 30 minutes	
Shield Building to Environment	6000±10%
Shield Building Recirculation	0.0
> 30 minutes	
Shield Building to Environment	3100
Shield Building Recirculation	2300

Table 12
Assumptions Used for Large Break LOCA Dose Analysis
(continued)

Spray Operation	
Time to initiate sprays	0.0 seconds
Time to terminate injection spray operation	0.91 hours
Time to establish recirculation spray	Not credited
Spray flow rates (gpm)	1300
Spray fall height (ft)	65
Removal Coefficients (hr^{-1})	
Injection spray elemental iodine removal	20.0
Injection spray particulate removal	5.0
Sedimentation particulate removal (after spray termination)	0.1
Credited sump volume (gal)	315000
ECCS leak rate to auxiliary building (gph)	6
Airborne fraction for ECCS leakage to auxiliary building (%)	
0 – 1.5 hours	10.0
> 1.5 hours	1.0
ECCS leak rate to RWST (gpm)	
0 – 24 hours	3.0
> 24 hours	1.5
Airborne fraction for ECCS leakage to RWST (%)	1.0
ECCS leakage plateout in auxiliary building (%)	50
Shield building and auxiliary building filter efficiencies (%)	
Elemental	90
Organic	90
Particulate	99
Time to start crediting emergency control room HVAC (min)	2

Table 13: Bounding Steam generator Tube Rupture Thermal Hydraulic Results
For Radiological Consequences Analysis

Reactor Trip Time (sec)	173.7
Loss of offsite power	At reactor trip
Tube rupture break flow (lbm)	
Pre-Trip	16,800
Post-Trip (until 30 minutes)	138,000
Tube rupture break flow flashing fraction (%)	
Pre-Trip	19.71
Post-Trip (until 30 minutes)	14.76
Steam Released to Environment	
Ruptured SG	
Pre-Trip (lbm/sec)	991.7
Post-Trip (until 30 minutes) (lbm)	85,000
Intact SG	
Pre-Trip (lbm/sec)	991.7
0 – 2 Hour (lbm)	243,800
2 – 8 Hour (lbm)	506,600

Table 14: Assumptions Used for Steam Generator Tube Rupture Dose Analysis

<u>Source Term</u>	
Coolant Activity	See Table 6
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)	4.0
<u>Release Modeling</u>	
Intact SG tube leak rate (gpd)	500
Thermal & hydraulic data	See Table 13
SG iodine water/steam partition coefficient	0.01
<u>Time to start crediting emergency control room HVAC (min)</u>	5

Table 15: Assumptions Used for Locked Rotor Dose Analysis

<u>Source Term</u>	
Core Activity	See Table 5
Fraction of fuel rods in core assumed to fail for dose considerations (% of core)	100
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Alkali Metals	12
Iodine chemical form in after release to atmosphere (%)	
Elemental	97
Organic	3
Particulate (cesium iodide)	0
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)	60
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 (gpd/SG)	500
Steam release to environment (lbm)	
0 – 2 Hours	213,000 +10%
2 – 8 hours	418,000 +10%
SG iodine and alkali metal water/steam partition coefficient	0.01
<u>Time to start crediting emergency control room HVAC (min)</u>	1

Table 16: Assumptions Used for Rod Ejection Dose Analysis

Source Term

Core Activity	See Table 5
Fraction of fuel rods in core that fail (% of core)	15
Gap Fractions (% of core activity)	
Iodine	10
Noble Gas	10
Alkali Metals	12
Fraction of fuel melting (% of core)	0.375
Radial peaking factor	1.7
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)	60
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 16
Assumptions Used for Rod Ejection Dose Analysis
(continued)

<u>Containment Leakage Release Path</u>

Containment net free volume (ft^3)	1.32E6
Shield building volume (ft^3)	3.74E5
Shield building participation fraction	0.5
Auxiliary building volume	Not Modeled, No holdup Credited
Containment leak rates (weight %/day)	
0 – 24 hours	0.5
> 24 hours	0.25
Containment Leak Path Fractions	
0-10 minutes	
Through Shield Building	0.0
Through Auxiliary Building SV	0.1
Direct to Environment	0.9
10 minutes - 30 days	
Through Shield Building	0.89
Through Auxiliary Building SV	0.10
Direct to Environment	0.01
Shield Building Air Flows (cfm)	
0-10 minutes	
Shield Building to Environment	Not applicable
Shield Building Recirculation	Not applicable
10 minutes – 30 minutes	
Shield Building to Environment	6000±10%
Shield Building Recirculation	0.0
> 30 minutes	
Shield Building to Environment	3100
Shield Building Recirculation	2300
Iodine chemical form in containment (%)	
Elemental	4.85
Organic	0.15
Particulate (cesium iodide)	95
Spray and sedimentation removal in containment	Not Credited
Shield building and auxiliary building filter efficiencies (%)	
Elemental	90
Organic	90
Particulate	99

Table 16
Assumptions Used for Rod Ejection Dose Analysis
(continued)

Primary to secondary Leakage Release Path	
SG tube leak rate (gpd/SG)	500
Steam release to environment (lbm/sec)	
0 – 200 seconds	800
200 – 1800 seconds	100
> 1800 seconds	0.0
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
Time to start crediting emergency control room HVAC (min)	2

Table 17: Assumptions Used for Fuel Handling Accident Dose Analysis

Radial peaking factor	1.7
Fuel damaged (number of assemblies)	1
Time from shutdown before fuel movement (hr)	100
Activity in the average fuel assembly after 100 hours shutdown (Ci)	
I-131	2.65E5
I-132	2.24E5
I-133	2.79E4
I-134	0.0
I-135	1.99E1
Kr-85m	1.98E-2
Kr-85	4.83E3
Kr-86	0.0
Kr-87	0.0
Xe-131m	3.99E3
Xe-133m	9.75E3
Xe-133	5.18E5
Xe-135m	3.19
Xe-135	9.80E2
Xe-138	0.0
Gap Fractions (% of core activity)	
I-131	8
Kr-85	10
Other Iodine and Noble Gas nuclides	5
Water depth	23 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 releases all activity (hours)	2
Time to start crediting emergency control room HVAC (min)	1

Table 18: 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)	4.0
SG tube leak rate (gpd/SG)	500
Steam release from faulted SG to environment during first two minutes (lbm)	150,000
Time to release initial mass in faulted SG (min)	2
Steam releases from intact SG (lbm)	
0 – 2 hours	220,000 +10%
2 – 8 hours	393,000 +10%
Time to cool RCS below 212°F and stop releases from faulted SG (hr)	72
SG iodine water/steam partition coefficient	
Faulted SG	1.0
Intact SG	0.01
Time to start crediting emergency control room HVAC (min)	5

Table 19: Assumptions Used for GDT and VCT Rupture Dose Analyses

Gas Decay Tank Rupture Source Term	
Gas decay tank inventory (Ci)	
Kr-85m	83
Kr-85	2480
Kr-87	15
Kr-88	184
Xe-131m	430
Xe-133m	558
Xe-133	34,300
Xe-135m	26
Xe-135	606
Xe-138	2
Volume Control Tank Rupture Source Term	
Volume control tank gas inventory (Ci)	
Kr-85m	62
Kr-85	637
Kr-87	15
Kr-88	232
Xe-131m	164
Xe-133m	257
Xe-133	14,000
Xe-135m	33
Xe-135	407
Xe-138	2
Volume control tank liquid iodine inventory (Ci)	
I-131	0.69
I-132	0.63
I-133	1.02
I-134	0.14
I-135	0.57
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)	60

Table 19
Assumptions Used for GDT and VCT Rupture Dose Analyses
(continued)

Release modeling	
Time to release all GDT activity (min)	5
Time to release all VCT activity (min)	5
Letdown flowrate to VCT (gpm)	80 +10%
Letdown demineralizer decontamination factor	10
Time to isolate letdown flow (min)	5
Time to start crediting emergency control room HVAC (min)	0.5