Enclosure 5 to AEP-NRC-2016-23

Red Wolf Associates (RWA) Technical Report

RWA-1313-015, Rev. 1, "D. C. Cook AST Radiological Analyses Technical Report"

Report Number: RWA-1313-015	Revision: 1

D. C. Cook AST Radiological Analyses Technical Report

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1 Introduction

The current D. C. Cook licensing basis for the radiological consequences analyses of accidents discussed in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) is based on methodologies prescribed in Reg. Guide 1.195 for the offsite doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ), and are based upon the Alternative Source Term (AST) methodology from Regulatory Guide 1.183 for the control room doses. These analyses are being updated to fully implement the AST methodology for both the onsite and offsite dose locations.

1.1 *Overview*

The revised design basis dose analyses are performed using the direction of Reg. Guide 1.183 (Reference [4.1]) with additional guidance provided in Regulatory Issues Summary (RIS) 2006-04 (Reference [4.2]). This single set of analyses is applicable to both Units 1 and 2, and as such, a limiting set of inputs are applied that is bounding for both units. In addition, releases from either unit must consider the dose impact on all receptor locations applicable to both units.

The following UFSAR Chapter 14 accidents are evaluated:

- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Locked Rotor
- Control Rod Ejection (CRE)
- Waste Gas Decay Tank (WGDT) Rupture
- Volume Control Tank (VCT) Rupture

It is important to note that Reg. Guide 1.183 does not include guidance for the analysis of the WGDT and VCT rupture accidents. These events are evaluated for completeness to apply a consistent source term to all of the radiological consequences presented in the D. C. Cook UFSAR and to assess the control room TEDE doses for these events against the acceptance criteria provided in 10CFR50.67. However, the offsite dose consequences resulting from the WGDT and VCT rupture events will continue to satisfy the acceptance criteria of 10CFR20 as discussed in Item 11 of Reference [4.2].

1.2 *Proposed Changes to the Licensing Basis*

As part of the full implementation of AST and the update of the D. C. Cook radiological dose analysis, the following changes to the Technical Specifications will be proposed.

- The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference Federal Guidance Report No. 11 as the source of effective dose conversion factors.
- Section 1.1 is revised to replace the definition of Ē Average Disintegration Energy with the definition of Dose Equivalent Xe-133 using dose conversion factors from the effective column of Table III.1 of Federal Guidance Report No. 12.
- The Limiting Condition for Operation related to RCS Activity in Section 3.4.16 is modified to replace the 100/Ē gross specific activity criterion with the Dose Equivalent Xe-133 limit.
- The accident induced leakage performance criterion established by the Steam Generator Program in Section 5.5.7.b.2 is revised to be 1 gpm for all steam generators and 0.25 gpm to any one steam generator.
- The maximum allowable leakage rate, L_a at P_a specified by the Containment Leakage Rate Program in Section 5.5.14.c is reduced to 0.18%/day.
- The maximum allowable methyl iodide penetration for the Control Room Emergency Ventilation charcoal adsorber is increased to 2.5% in Section 5.5.9.c.

1.3 *Computer Codes*

The following computer codes are used in performing the Cook radiological dose analyses:

Computer Code	Version	Reference
RADTRAD	3.10	[4.3] - [4.6]
ARCON96	1997	[4.7]
PAVAN	2.0	[4.8]
JFREQ (METD)	1982	[4.29]
MicroShield	8.03	[4.10]
ORIGEN-ARP	6.1.3	[4.9]
GOTHIC	7.2a	[4.11] - [4.13]

RADTRAD is used to determine the control room and offsite doses for each analyzed event using the source term and *X/Q* inputs. The code considers the release timing, filtration, hold-up, and chemical form of the nuclides released into the environment.

ARCON96 is used to determine the atmospheric dispersion factors (*X/Qs*) at the control room intakes for selected release locations from plant meteorological data.

PAVAN provides atmospheric dispersion factors (*X/Qs*) for various time periods at the EAB and LPZ boundaries using plant meteorological data.

JFREQ is a program in the METD suite of programs that is used to compute the joint frequency distribution of wind speed, wind direction, and atmospheric stability class for use as input to the PAVAN program.

MicroShield is used to determine the direct shine dose to the operators in the control room from the activity on the control room ventilation system filters.

ORIGEN-ARP calculates the fission product isotopic activity of the reactor core used in the development of the core and RCS source terms.

The GOTHIC code is used to simulate the RCS purification system to determine the relative concentrations of nuclides in the reactor coolant, and is also used to calculate the time-dependent RWST temperature due to backleakage from the containment sump.

2 Common Input Parameters, Assumptions, and Methods

2.1 *Control Room Dose Calculation Model*

During normal operation, 880 cfm of unfiltered air enters the control room through the normal outside air intake. Following a safety injection signal, the control room ventilation system is automatically placed into recirculation after applicable delays for signal processing, emergency power restoration, and damper repositioning. In this configuration, the control room pressurization/cleanup fans circulate 5400 cfm of air through the control room filters, with 880 cfm of this flow supplied by fresh air from the emergency outdoor air intake and the remaining 4520 cfm taken from the control room envelope.

Unfiltered inleakage is assumed to enter the control room at a constant rate of 40 cfm during all modes of system operation. When the control room ventilation system is aligned in the pressurization/cleanup mode, the control room envelope is at a positive pressure with respect to the surrounding areas and leakage is predominantly out of the control room. However, this flow configuration creates a negative pressure in the system ducting downstream of the isolated normal intake dampers. Therefore, the control room unfiltered inleakage is assumed to enter the control room at the location of the normal intakes. Control room occupancy and breathing rates are taken from Position 4.2.6 of Reference [4.1]. The control room model parameters used in the analyses are listed in Table 2.1-1.

Table 2.1-1: Control Room Parameters

Parameter	Value
Control Room Volume	50,616 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	880 cfm
Unfiltered Inleakage	40 cfm
Emergency Operation	
Recirculation Mode:	
Filtered Make-up Flow Rate	880 cfm
Filtered Recirculation Flow Rate	4520 cfm
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage	40 cfm
Filter Efficiencies	
Elemental	94.05%
Organic	94.05%
Particulate	98.01%
Occupancy	
0 – 24 hrs 1 – 4 days 4 – 30 days	1.0 0.6 0.4
Breathing Rate	3.5×10^{-4} m^3/sec

2.2 *Source Terms*

2.2.1 Core Source Term

Consistent with the guidance of Position 3.1 of Reference [4.1], the inventory of fission products in the reactor core available for release is based upon the product of the maximum full power operation of the core at the licensed rated thermal power and the ECCS evaluation uncertainty. An ORIGEN-ARP model is created using a single fuel assembly as the mass basis with an assembly burnup selected which conservatively exceeds the expected end-of-cycle core average burnup. Separate ORIGEN-ARP cases are run covering the full range of licensed fuel enrichments, and the maximum activity for each dose-significant isotope is selected from among

these cases. In this manner, the fission product activity of a single assembly is derived which bounds the anticipated core design values for enrichment and burnup. The total core average source term is then obtained by multiplying the single-assembly isotopic activities by the number of fuel assemblies in the core. Key source term inputs are presented in Table 2.2-1.

The list of dose significant isotopes is generally based upon Table 5 of Reference [4.1], which identifies the radionuclide elements that should be considered in the design basis AST dose analysis. This element list is the same as that specified for the revised source term in Table 3.8 of NUREG-1465 (Reference [4.14]). Section 1.4.3.2 of Reference [4.4] documents the creation of a 60-isotope, 9-element list of radionuclides which meets the requirements of NUREG-1465. However, some early AST submittals included a more comprehensive list of isotopes. Specifically, Table 3.1-4 of Reference [4.15] presents a core inventory which contains the standard 60 isotopes from Table 1.4.3.2-2 of Reference [4.4] plus an additional 61 nuclides. This extended list serves as the refined basis for the dose-significant isotopes selected for this analysis; however, the number of isotopes is reduced to a maximum of 100 to accommodate the limitations of RADTRAD 3.10 identified in Section 1.1 of Reference [4.3]. The reduction from 121 to 100 nuclides is based upon the availability of valid dose conversion factors from References [4.16] and [4.17], nuclide half life, and significance of daughter products. The final list of the 100 dose significant isotopes and corresponding core source term activities is presented in Table 2.2-2. This source term is used in combination with dose conversion factors from Federal Guidance Report No. 11 (FGR 11) (Reference [4.16]) and Federal Guidance Report No. 12 (FGR 12) (Reference [4.17]), which is consistent with Position 4.1 of Reference [4.1].

2.2.2 Fuel Handling Accident Source Term

The fuel handling accident source term is developed from the core source term and follows the guidance of Position 3.1 of Reference [4.1], which states that the fission product inventory of each damaged fuel rod for DBA events that do not involve the entire core is determined by dividing the total core inventory by the number of rods in the core. To account for differences in power level across the core, the radial peaking factor is applied to the inventory of the damaged rods. Since the fuel handling accident event involves the failure of all of the rods in a single fuel assembly, the FHA source term is derived by dividing each nuclide activity in the core source term by the number of fuel assemblies and multiplying by the radial peaking factor.

Two additional adjustments are needed to finalize the FHA source term. The gap inventory fractions for I-131 and Kr-85 specified in Table 3 of Reference [4.1] are greater than the respective halogen and noble gas gap fractions. To accommodate these differences, the activities of these two isotopes are increased so that the proper quantities are released when the group gap fractions are applied. Specifically, the activity of I-131 is increased by a factor of 0.08/0.05 = 1.6, and the activity of Kr-85 is increased by 0.1/0.05 = 2.0. Thus, using the core average activities from Table 2.2-2, the radial peaking factor of 1.65 from Table 2.2-1, and 193 fuel assemblies in the core, the fuel handling source term shown in Table 2.2-3 is obtained.

2.2.3 RCS Source Term

The equilibrium nuclide concentration in the RCS is calculated based on the core inventory described in Section 2.2.1. The rate of nuclide release from the core to the reactor coolant for applicable isotopes is calculated from fission product escape rate coefficients and assumes that 1% of fuel rods have defects. With this isotopic production rate in the coolant established, RCS concentrations are calculated with a hydraulic

model of the RCS purification system using GOTHIC. This model accounts for radioactive decay and daughter production, removal of nuclides by the demineralizers, degassing in the volume control tank (VCT), and dilution of the nuclide concentration by normal makeup for RCS boron control. The GOTHIC model is run until the radionuclide concentrations reach equilibrium values. The GOTHIC output provides the relative distribution of isotopic concentrations in the RCS. These values are then manually scaled such that the iodine activities match the Dose Equivalent I-131 limit of 1.0 μCi/gm specified in the Technical Specifications, and the non-iodine isotopes are adjusted separately to meet the gross specific activity limits of 100/Ē. The resulting RCS equilibrium source term is shown in Table 2.2-4. The noble gas concentrations from Table 2.2-4 correspond to a Dose Equivalent Xe-133 of 215.1 μCi/gm based upon the definition of Dose Equivalent Xe-133 provided in TSTF-490 (Reference [4.30]).

2.2.4 Steam Generator Secondary Source Term

The specific iodine activity on the secondary side of the steam generators is limited to 0.1 μCi/gm Dose Equivalent I-131, as shown in Table 2.2-1, which is one-tenth of the RCS iodine activity limit. Therefore, the secondary activities are developed by taking the iodine activities from Table 2.2-4 and reducing them by a factor of ten. The resulting secondary source term is presented in Table 2.2-5.

Table 2.2-1: Source Term Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3480 MWt (3468 MWt plus 0.34% uncertainty)
Core Average Fuel Burnup	43,000 MWD/MTU
Fuel Enrichment	1.5 –5.0 w/o
Uranium Mass per Fuel Assembly	498 kg
Number of Assemblies in the Core	193
Assembly Radial Peaking Factor	1.65
RCS Iodine Specific Activity Limit	1.0 μCi/gm Dose Equivalent I-131
SG Secondary Iodine Specific Activity Limit	0.1 μCi/gm Dose Equivalent I-131
Non-Iodine Gross Specific Activity Limit	100/Ē

Table 2.2-2: Core Source Term

Nuclide	Activity (Curies)	Nuclide	Activity (Curies)
Co-58	8.884E+05	Pr-143	1.398E+08
Co-60	6.796E+05	Nd-147	6.178E+07
Kr-85	1.280E+06	Np-239	2.609E+09
Kr-85m	2.364E+07	Pu-238	4.130E+05
Kr-87	4.661E+07	Pu-239	3.727E+04
Kr-88	6.222E+07	Pu-240	6.637E+04
Rb-86	2.272E+05	Pu-241	1.603E+07
Sr-89	8.677E+07	Am-241	1.707E+04
Sr-90	1.002E+07	Cm-242	7.417E+06
Sr-91	1.100E+08	Cm-244	1.838E+06
Sr-92	1.184E+08	Kr-83m	1.119E+07
Y-90	1.038E+07	Br-82	3.972E+05
Y-91	1.142E+08	Br-83	1.106E+07
Y-92	1.197E+08	Br-84	2.009E+07
Y-93	1.358E+08	Rb-89	8.303E+07
Zr-95	1.566E+08	Y-91m	6.384E+07
Zr-97	1.586E+08	Y-95	1.496E+08
Nb-95	1.578E+08	Nb-95m	1.795E+06
Mo-99	1.742E+08	Nb-97	1.596E+08
Tc-99m	1.546E+08	Rh-103m	1.849E+08
Ru-103	1.850E+08	Pd-109	5.749E+07
Ru-105	1.491E+08	Sb-124	1.434E+05
Ru-106	9.480E+07	Sb-125	1.231E+06
Rh-105	1.309E+08	Sb-126	5.873E+04
Sb-127	1.067E+07	Te-125m	2.725E+05
Sb-129	3.215E+07	Te-131	8.174E+07
Te-127	1.054E+07	Te-133	1.024E+08
Te-127m	1.841E+06	Te-133m	8.990E+07
Te-129	3.017E+07	Te-134	1.700E+08
Te-129m	5.821E+06	I-130	3.945E+06
Te-131m	2.119E+07	Xe-131m	1.385E+06
Te-132	1.374E+08	Xe-133m	6.099E+06
I-131	9.814E+07	Xe-135m	4.335E+07
I-132	1.420E+08	Xe-138	1.627E+08

Nuclide	Activity (Curies)	Nuclide	Activity (Curies)
I-133	1.916E+08	Cs-134m	5.865E+06
I-134	2.148E+08	Cs-138	1.776E+08
I-135	1.832E+08	Ba-141	1.522E+08
Xe-133	1.919E+08	La-143	1.419E+08
Xe-135	5.900E+07	Pm-147	1.944E+07
Cs-134	2.523E+07	Pm-148	1.841E+07
Cs-136	6.388E+06	Pm-148m	4.711E+06
Cs-137	1.325E+07	Pm-149	6.245E+07
Ba-139	1.693E+08	Pm-151	2.177E+07
Ba-140	1.639E+08	Sm-153	6.797E+07
La-140	1.700E+08	Eu-154	9.557E+05
La-141	1.533E+08	Eu-155	4.427E+05
La-142	1.475E+08	Eu-156	4.798E+07
Ce-141	1.548E+08	Np-238	5.165E+07
Ce-143	1.430E+08	Pu-243	1.153E+08
Ce-144	1.296E+08	Am-242	1.148E+07

Table 2.2-3: Fuel Handling Accident Source Term

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Co-58	7.595E+03	Pr-143	1.195E+06
Co-60	5.810E+03	Nd-147	5.282E+05
Kr-85	2.189E+04	Np-239	2.230E+07
Kr-85m	2.021E+05	Pu-238	3.531E+03
Kr-87	3.985E+05	Pu-239	3.186E+02
Kr-88	5.319E+05	Pu-240	5.674E+02
Rb-86	1.942E+03	Pu-241	1.370E+05
Sr-89	7.418E+05	Am-241	1.459E+02
Sr-90	8.566E+04	Cm-242	6.341E+04
Sr-91	9.404E+05	Cm-244	1.571E+04
Sr-92	1.012E+06	Kr-83m	9.567E+04
Y-90	8.874E+04	Br-82	3.396E+03
Y-91	9.763E+05	Br-83	9.455E+04
Y-92	1.023E+06	Br-84	1.718E+05

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Y-93	1.161E+06	Rb-89	7.098E+05
Zr-95	1.339E+06	Y-91m	5.458E+05
Zr-97	1.356E+06	Y-95	1.279E+06
Nb-95	1.349E+06	Nb-95m	1.535E+04
Mo-99	1.489E+06	Nb-97	1.364E+06
Tc-99m	1.322E+06	Rh-103m	1.581E+06
Ru-103	1.582E+06	Pd-109	4.915E+05
Ru-105	1.275E+06	Sb-124	1.226E+03
Ru-106	8.105E+05	Sb-125	1.052E+04
Rh-105	1.119E+06	Sb-126	5.021E+02
Sb-127	9.122E+04	Te-125m	2.330E+03
Sb-129	2.749E+05	Te-131	6.988E+05
Te-127	9.011E+04	Te-133	8.754E+05
Te-127m	1.574E+04	Te-133m	7.686E+05
Te-129	2.579E+05	Te-134	1.453E+06
Te-129m	4.977E+04	I-130	3.373E+04
Te-131m	1.812E+05	Xe-131m	1.184E+04
Te-132	1.175E+06	Xe-133m	5.214E+04
I-131	1.342E+06	Xe-135m	3.706E+05
I-132	1.214E+06	Xe-138	1.391E+06
I-133	1.638E+06	Cs-134m	5.014E+04
I-134	1.836E+06	Cs-138	1.518E+06
I-135	1.566E+06	Ba-141	1.301E+06
Xe-133	1.641E+06	La-143	1.213E+06
Xe-135	5.044E+05	Pm-147	1.662E+05
Cs-134	2.157E+05	Pm-148	1.574E+05
Cs-136	5.461E+04	Pm-148m	4.028E+04
Cs-137	1.133E+05	Pm-149	5.339E+05
Ba-139	1.447E+06	Pm-151	1.861E+05
Ba-140	1.401E+06	Sm-153	5.811E+05
La-140	1.453E+06	Eu-154	8.170E+03
La-141	1.311E+06	Eu-155	3.785E+03
La-142	1.261E+06	Eu-156	4.102E+05
Ce-141	1.323E+06	Np-238	4.416E+05
Ce-143	1.223E+06	Pu-243	9.857E+05
Ce-144	1.108E+06	Am-242	9.815E+04

Table 2.2-4: RCS Source Term

Nuclide	Activity (μCi/g)	Nuclide	Activity (μCi/g)
Co-58	0.000E+00	Pr-143	6.713E-03
Co-60	0.000E+00	Nd-147	0.000E+00
Kr-85	2.385E+01	Np-239	0.000E+00
Kr-85m	5.204E-01	Pu-238	0.000E+00
Kr-87	3.299E-01	Pu-239	0.000E+00
Kr-88	9.148E-01	Pu-240	0.000E+00
Rb-86	8.797E-02	Pu-241	0.000E+00
Sr-89	1.335E-03	Am-241	0.000E+00
Sr-90	1.237E-04	Cm-242	0.000E+00
Sr-91	5.681E-04	Cm-244	0.000E+00
Sr-92	2.488E-04	Kr-83m	1.350E-01
Y-90	2.152E-04	Br-82	4.641E-03
Y-91	1.692E-02	Br-83	2.720E-02
Y-92	3.067E-04	Br-84	1.244E-02
Y-93	2.010E-04	Rb-89	2.530E-02
Zr-95	2.409E-02	Y-91m	3.314E-04
Zr-97	3.920E-04	Y-95	0.000E+00
Nb-95	3.478E-02	Nb-95m	1.867E-04
Mo-99	2.070E+00	Nb-97	4.900E-05
Tc-99m	1.980E+00	Rh-103m	1.988E-02
Ru-103	1.991E-02	Pd-109	0.000E+00
Ru-105	9.723E-05	Sb-124	0.000E+00
Ru-106	3.340E-02	Sb-125	0.000E+00
Rh-105	7.689E-04	Sb-126	0.000E+00
Sb-127	0.000E+00	Te-125m	2.449E-02
Sb-129	0.000E+00	Te-131	1.599E-02
Te-127	2.489E-01	Te-133	0.000E+00
Te-127m	2.465E-01	Te-133m	7.643E-03
Te-129	2.281E-01	Te-134	1.092E-02
Te-129m	3.463E-01	I-130	0.000E+00
Te-131m	5.787E-02	Xe-131m	1.600E+00
Te-132	9.639E-01	Xe-133m	1.423E+00
I-131	8.087E-01	Xe-135m	2.138E-01
I-132	6.411E-01	Xe-138	2.292E-01

Nuclide	Activity (µCi/g)	Nuclide	Activity (µCi/g)
I-133	1.0304E+00	Cs-134m	2.031E-02
I-134	1.231E-01	Cs-138	3.420E-01
I-135	5.365E-01	Ba-141	4.233E-05
Xe-133	1.037E+02	La-143	0.000E+00
Xe-135	3.361E+00	Pm-147	0.000E+00
Cs-134	3.327E+01	Pm-148	0.000E+00
Cs-136	2.188E+00	Pm-148m	0.000E+00
Cs-137	1.852E+01	Pm-149	0.000E+00
Ba-139	1.975E-04	Pm-151	0.000E+00
Ba-140	1.940E-03	Sm-153	0.000E+00
La-140	2.878E-03	Eu-154	0.000E+00
La-141	1.301E-04	Eu-155	0.000E+00
La-142	3.346E-05	Eu-156	0.000E+00
Ce-141	1.445E-02	Np-238	0.000E+00
Ce-143	6.911E-04	Pu-243	0.000E+00
Ce-144	4.229E-02	Am-242	0.000E+00

Table 2.2-5: SG Secondary Source Term

Nuclide	Activity (µCi/g)
I-131	8.087E-02
I-132	6.411E-02
I-133	1.0304E-01
I-134	1.231E-02
I-135	5.365E-02

2.2.5 Fuel Rod Gap Fractions and High Burnup Rods

The fraction of the core fission product inventory located within the fuel rod gap for non-LOCA events is specified in Table 3 of Reference [4.1] and shown below. These values apply to fuel rods that are damaged as a result of the event, and the entire contents of the fuel rod gap are instantaneously released from the fuel. Note that separate gap inventory fractions of 10% for both noble gases and iodines are applied in the Control Rod Ejection event as required by Position 1 of Appendix H to Reference [4.1].

Table 2.2-6: Non-LOCA Fuel Rod Gap Inventory Fraction

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

Footnote 11 of Reference [4.1] states that these fuel rod gap inventories are acceptable for fuel rods with a peak burnup of up to 62,000 MWD/MTU provided that the linear heat generation rate does not exceed 6.3 kw/ft for rods with burnups greater than 54 GWD/MTU. For this analysis, a total of 150 rods in two fuel assemblies are assumed to exceed the burnup limits of Footnote 11 to allow for future core design margin. Similar high burnup concerns were raised in alternative source term submittals by Fort Calhoun, Byron/Braidwood, and St. Lucie stations. For these plants, the issue was addressed by doubling the gap fractions for 100% of the rods in the affected assemblies and applying the maximum radial peaking factor. This approach is discussed in Section 2.2.4 of References [4.18], Section 3.3 of Reference [4.19], and Section 2.9.2.2.2.1 of References [4.20] and [4.21]. This same methodology for addressing high burnup fuel is applied to this analysis.

2.3 *Atmospheric Dispersion Factors*

The dose analysis addresses releases from either unit and must consider the dose impact on all receptor locations applicable to both units. As such, atmospheric dispersion factors are developed for all possible release-receptor pairs, and the values applied in the analysis reflect the most limiting combination without regard to the unit in which the event occurs.

2.3.1 Onsite *X/Q* Determination

New *X/Q* factors for onsite release-receptor combinations are developed using the ARCON96 computer code (NUREG/CR-6331, Reference [4.7]). Reg. Guide 1.194 (Reference [4.26]) contains new guidance that supersedes the NUREG/CR-6331 recommendations for using certain default parameters as input. Therefore, the following changes from the default values are made:

- For surface roughness length, m, a value of 0.2 is used in lieu of the default value of 0.1, and

- For averaging sector width constant, a value of 4.3 is used in lieu of the default value of 4.0.

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

Figure 2.3-4 provides a sketch of the general layout of the Cook plant that has been annotated to highlight the onsite release and receptor point locations. All releases are taken as ground level releases per the guidance of Position 3.2.1 of Reg. Guide 1.194.

Table 2.3-2 provides information related to the relative elevations of the release-receptor combinations, the straight-line horizontal distance between the release point and the receptor location, and the direction (azimuth) from the receptor location to the release point. Angles are calculated based on trigonometric layout of release and receptor points in relation to the North-South and East-West axes. Plant North is offset 16.39 degrees clockwise from True North (17° 55' 40.5" - 1° 32' 32" = 16.39°) as shown in Figure 2.3-3. This offset is taken into account in development of the release-receptor pair angles.

A building wake term is only applied to releases close to or on the containment surface, where it is clear that the effect of the containment building wake will influence the result. Such releases include the Containment Vent, Containment Surface, Western SG PORVS/MSSVs, West Main Steam Enclosure, and Containment Diffuse Area locations for Units 1 and 2. The building area used for this wake term is 1,690 m^2.

For scenarios with diffuse area releases, the building surface is modeled as a vertical planar area (or diffuse area) source in ARCON96 via two initial diffusion coefficients. The coefficients are calculated based on the guidance found in Position 3.2.4.4 of Reg. Guide 1.194. The area source width is the containment diameter of approximately 122 feet (ft), resulting in an initial diffusion coefficient σ_{Y_0} of 6.2 m. The area source height, based conservatively on the elevation difference between the top of the containment building and the top of the tallest building connected to the containment surface, is approximately 89.5 ft, resulting in an initial diffusion coefficient σ_{Z_0} of 4.55 m.

Table 2.3-3 provides the Control Room *X/Q* factors for the release-receptor combinations. These factors are not corrected for occupancy. This table summarizes the *X/Q* factors for the control room intakes used in the various accident scenarios for onsite control room dose consequence analyses. Values are presented for the normal and emergency intakes for each unit.

Table 2.3-7 identifies the Release-Receptor pair and associated Control Room *X/Q* factors from Table 2.3-3 that are used in the event analyses during each of the modes of control room ventilation.

2.3.2 Offsite *X/Q* Determination

For offsite receptor locations, the new atmospheric dispersion (*X/Q*) factors are developed using the PAVAN computer code (NUREG/CR-2858, Reference [4.8]). Table 2.3-4 provides the minimum distance to the EAB and LPZ in each direction for each release location. The offsite maximum *X/Q* factors for the EAB and LPZ are presented in Table 2.3-6. According to Regulatory Position 4 of Reg. Guide 1.145 (Reference [4.27], the dose calculations should use the larger value between the maximum downwind sector and the 5% overall site *X/Q* values for each boundary and time period. For each individual PAVAN case evaluated, the 5% overall site *X/Q* values are bounded by the maximum sector *X/Q*. Therefore, the *X/Q* values for use in the EAB and LPZ dose evaluations for all releases are based on the maximum sector values provided in Table 2.3-6.

Offsite release-receptor pair locations are illustrated in Figure 2.3-5. The application of the X/Q values from Table 2.3-6 in the respective event analyses is identified in Table 2.3-7. All of the releases are considered to

be ground level releases based upon Position 1.3.2 of Reg. Guide 1.145. As such, the release height in PAVAN is set equal to 10.0 meters as required by Table 3.1 of NUREG/CR-2858. The building area used for the building wake term is the same as for some of the ARCON96 onsite *X/Q* cases, which is 1,690 m^2. The building height entered into PAVAN is the top elevation of the containment dome (49.5 m).

2.3.3 Meteorological Data

Meteorological data from a five-year data set (2002, 2004, 2005, 2007, 2010) is used in the development of the new onsite and offsite *X/Q* factors used in the analysis. These years were selected since they were the five most recent years with full periods of high quality data available (i.e., the data set from each year has greater than 90% of the possible hours recovered and used as input to the analyses). The meteorological data is converted from the raw format into the proper formatting required to create the meteorological data files for use with ARCON96 and PAVAN. The Cook Nuclear Plant records meteorological data from a primary, backup, and shoreline tower.

The data from the shoreline tower is considered to most accurately represent the meteorological conditions on-site based on its vicinity to the plant. However, the shoreline tower only records data at a height of 10 meters. As a result, for on-site analyses, a hybrid meteorological data set is created using the 10 meter shoreline data for the lower level measurements and data from the 60 meter level on the primary tower for the upper level measurements. The stability classes are calculated based on the temperature difference between the 10 m and 60 m levels on the primary tower. The use of stability classes based on the primary tower data is considered to be conservative because the primary tower likely experiences more stable conditions due to its location further inland.

For off-site analyses, the use of primary tower data versus the shoreline tower data varies based on downwind sectors. A ridgeline running north-south is located approximately 0.5 miles east of the plant site. For locations west of the ridgeline, the shoreline tower data is more representative. For locations east of the ridgeline, the primary tower data is more representative. As a result, both the primary tower data and shoreline tower data is formatted into the joint frequency distribution for use in off-site analyses. Similar to the on-site analysis, the shoreline tower distribution is based on a hybrid meteorological data set using the 10 meter shoreline tower data for the lower level measurements, the 60 meter primary tower data for the upper level measurements, and stability classes calculated based on the temperature readings from the primary tower.

Due to the multiple joint frequency distributions, two PAVAN cases are developed for each release location, with one case incorporating the primary tower distribution and the other incorporating the shoreline tower distribution. The maximum *X/Q* for each downwind direction at the EAB and LPZ is taken from the appropriate PAVAN output (shoreline or primary) based on the sector's location relative to the ridgeline described above. The correlation between downwind direction and representative meteorological data set is provided in Table 2.3-5. Figure 2.3-1 and Figure 2.3-2 also present this information graphically.

Figure 2.3-1: EAB Downwind Sectors

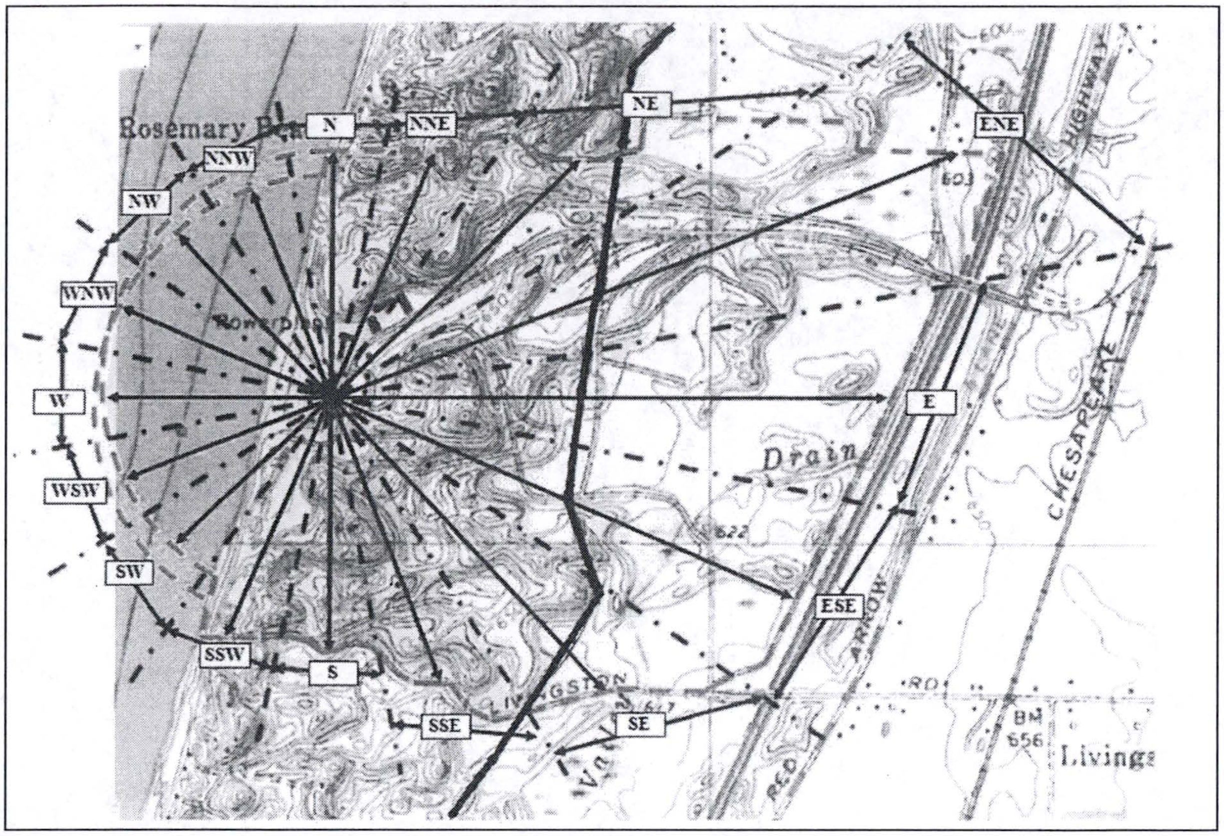
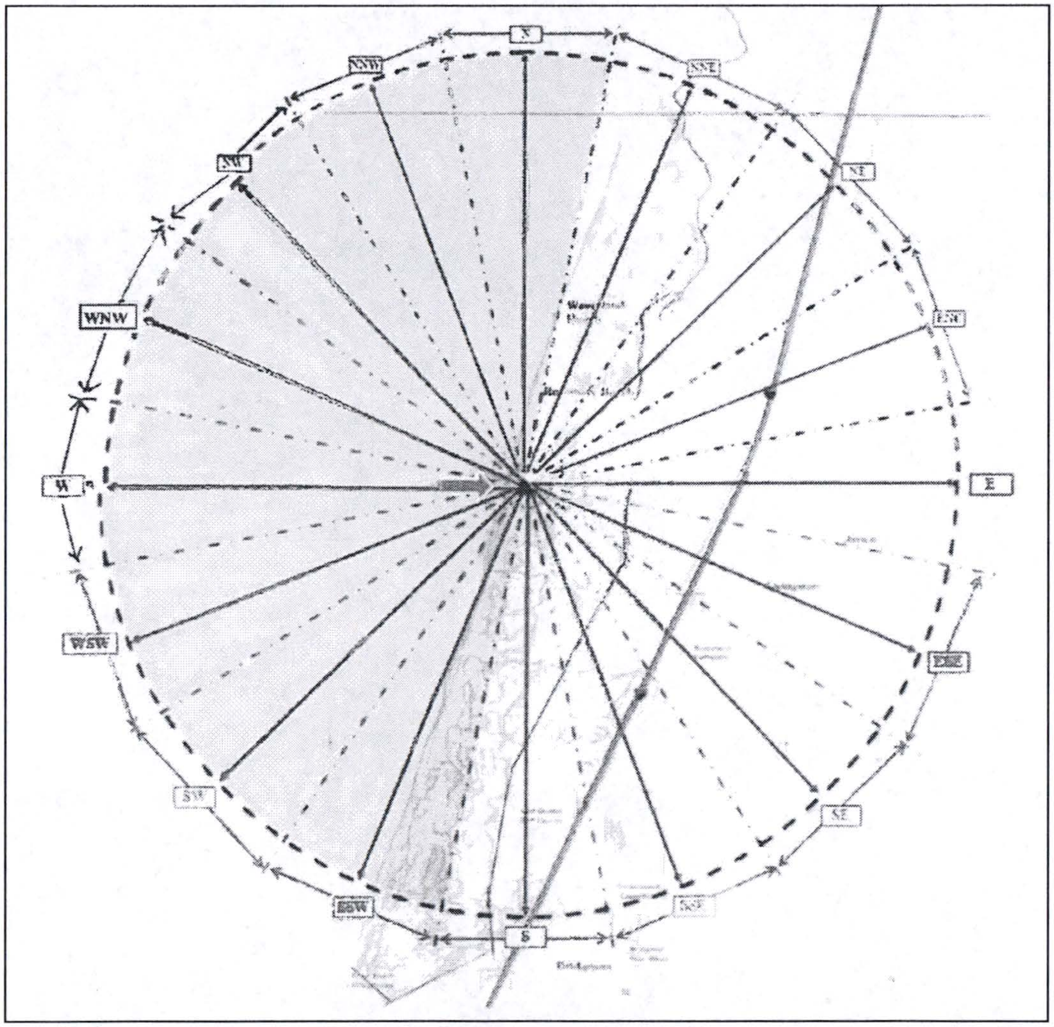


Figure 2.3-2: LPZ Downwind Sectors



The years of data provided are based on the 5 most recent years that have valid data for both the primary and shoreline towers. Five years worth of meteorological data is used which meets the guidance set forth in Regulatory Position 3.1 of Reg. Guide 1.194. The raw data for the years listed above was manipulated within a spreadsheet for appropriate formatting for use with ARCON96 and PAVAN.

The raw data was examined to identify and flag bad or missing data to ensure that the meteorological data used in the atmospheric dispersion factor determination were of high quality. All bad or missing data was removed from the data set prior to use in ARCON96 and PAVAN. No regulatory guidance is provided in Reg. Guide 1.194 and NUREG/CR-6331 on the valid meteorological data recovery rate required for use in determining onsite *X/Q* values. However, Regulatory Position 5 of Reg. Guide 1.23 (Reference [4.28]) specifies a 90%

data recovery threshold for measuring and capturing meteorological data. The 90% data recovery rate applies to the composite of all variables needed to model atmospheric dispersion for each potential release pathway. For the 2002, 2004, 2005, 2007, and 2010 data base, the meteorological data recovery rates are listed in Table 2.3-1.

Table 2.3-1: Meteorological Data Recovery Rate

Parameter	2002	2004	2005	2007	2010
	%	%	%	%	%
Wind Speed 10m Primary	99.9	99.4	99.8	99.6	99.4
Wind Speed 60m Primary	99.9	99.4	99.8	94.3	89.5
Wind Speed 10m Shore	100	96	99.8	99.9	99.2
Wind Direction 10m Primary	99.9	99.4	99.8	99.6	99.5
Wind Direction 60m Primary	99.9	99.4	99.8	99.6	91.8
Wind Direction 10m Shore	100	96	99.8	100	98.6
Delta Temperature 60-10m	99.8	99.3	99.6	99.4	98.1

While the 60 m wind speed for 2010 has a recovery of 89.5%, it is considered acceptable as the cumulative recovery rate for all years of 60 m wind speeds and all parameters for 2010 are well above 90%. With a total of five years worth of data, the contents of the meteorological data file are representative of the long-term meteorological trends at the Cook site.

The raw meteorological data was also processed into annual joint frequency distribution format for 2002, 2004, 2005, 2007, and 2010 for the offsite atmospheric dispersion factor analysis. The joint frequency distribution file requires the annual meteorological data to be sorted into three classifications that include wind direction, wind speed, and atmospheric stability class. The format for the file conforms to the format provided in Table 3 of Reg. Guide 1.23. The data for all years was sorted into wind speed bins using the guidance provided in RIS 2006-04.

The PAVAN code requires that the maximum speed for each wind speed category be input. The guidance provided in RIS 2006-04 gives a maximum wind speed category of 10 m/s. However, in order to be consistent with Cook meteorological data evaluations, an upper limit of 14 m/s is chosen in order to capture the average of all winds greater than 10 m/s (approximately 12 m/s) from the shoreline tower. The primary tower data set only has 2 hours of data with wind speeds higher than 10 m/s, which are included in the highest wind speed bin of 14 m/s.

Figure 2.3-3: Site Orientation

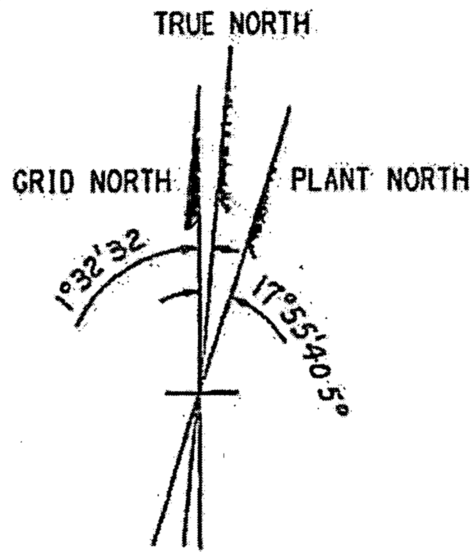
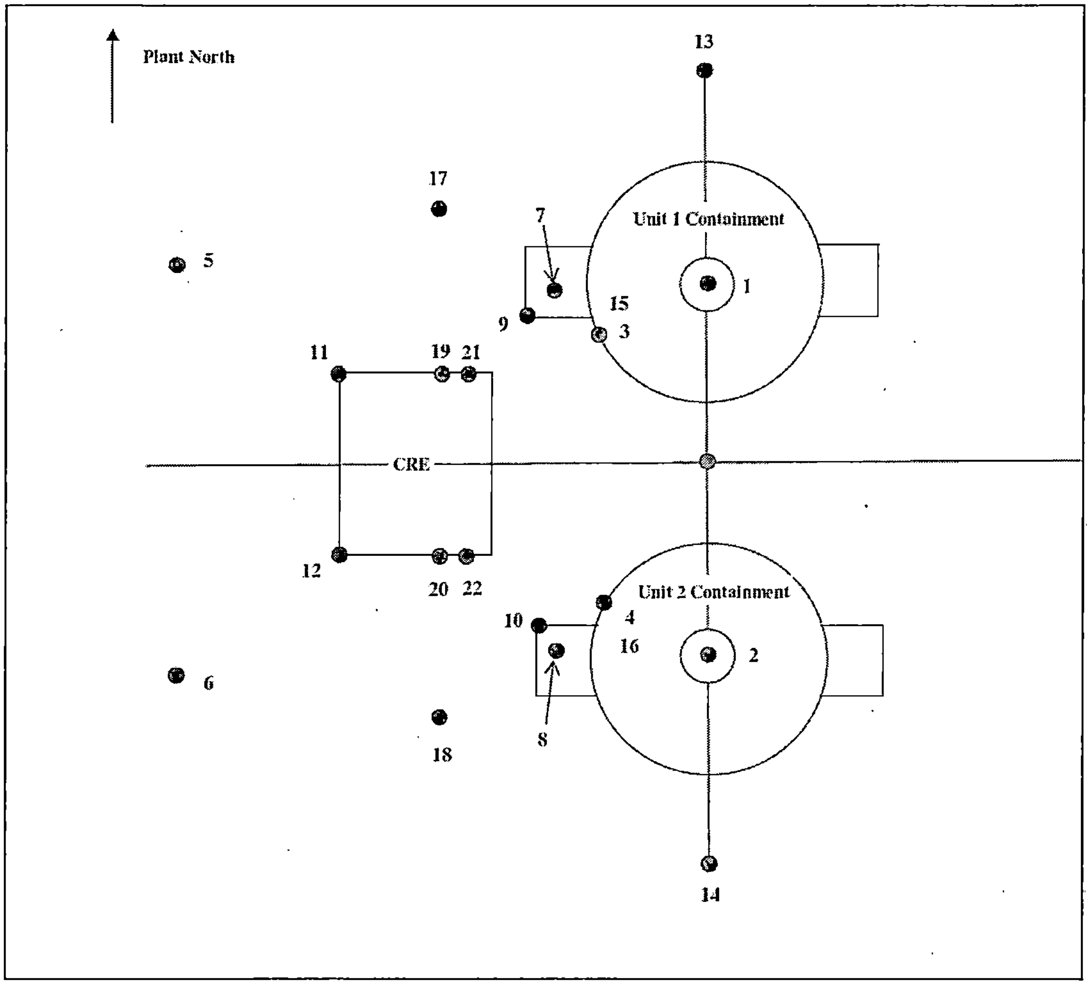


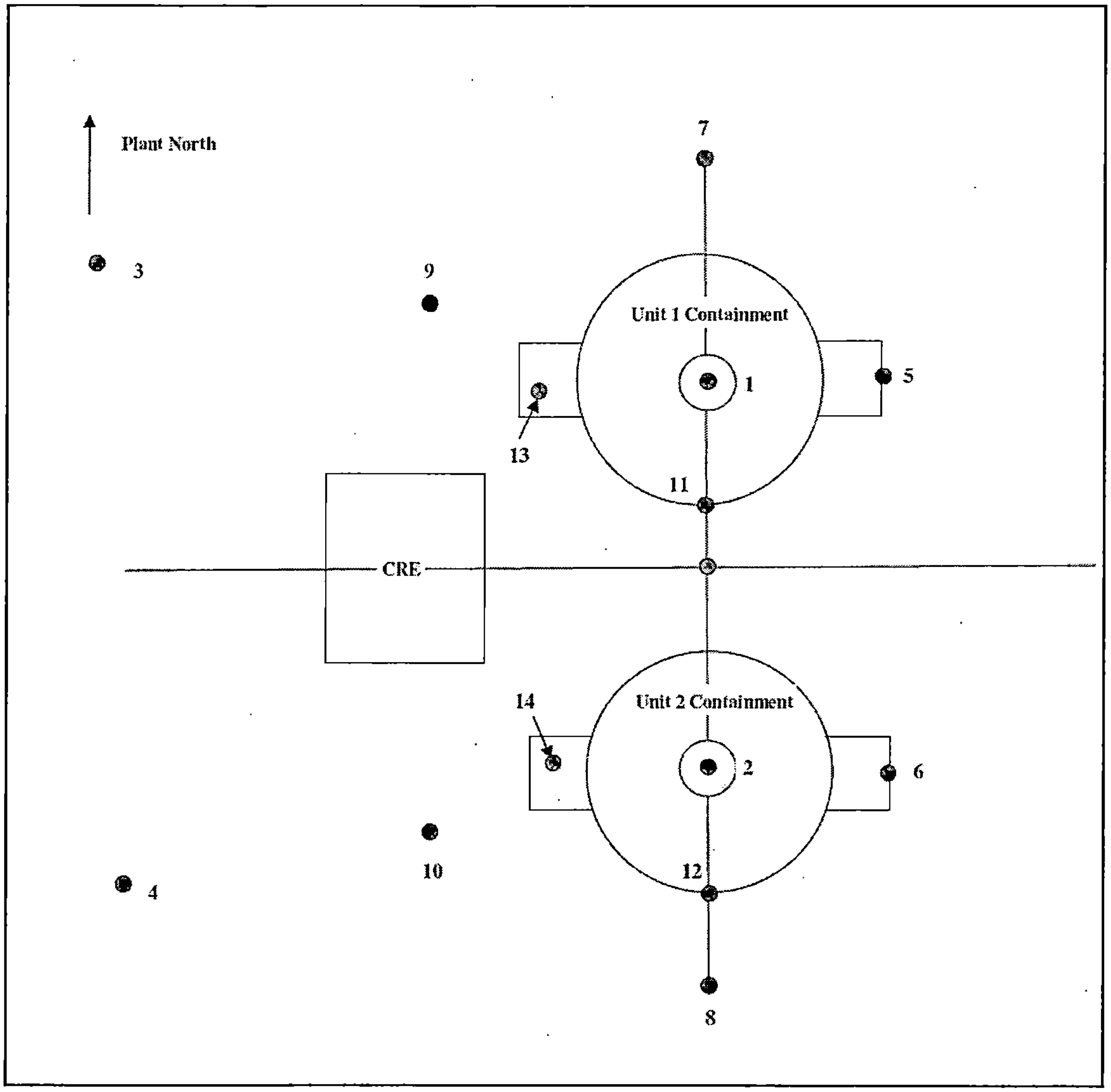
Figure 2.3-4: Onsite Release-Receptor Locations



* Figure is not to scale

1 - Unit 1 Containment Vent
2 - Unit 2 Containment Vent
3 - Closest Point on Unit 1 Containment
4 - Closest Point on Unit 2 Containment
5 - Unit 1 Steam Jet Air Injectors
6 - Unit 2 Steam Jet Air Injections
7 - Unit 1 Western SG PORVs/MSSVs
8 - Unit 2 Western SG PORVs/MSSVs
9 - Unit 1 West Main Steam Enclosure
10 - Unit 2 West Main Steam Enclosure
11 - Unit 1East Turbine Building
12 - Unit 2 East Turbine Building
13 - Unit 1 RWST
14 - Unit 2 RWST
15 - Unit 1 Containment Diffuse Area
16 - Unit 2 Containment Diffuse Area
17 - North Auxiliary Building Supply
18 - South Auxiliary Building Supply
19 - Unit 1 Normal Control Room Intake
20 - Unit 2 Normal Control Room Intake
21 - Unit 1 Emergency Control Room Intake
22 - Unit 2 Emergency Control Room Intake

Figure 2.3-5: Offsite Release-Receptor Locations



* Figure is not to scale

1 - Unit 1 Containment Vent
2 - Unit 2 Containment Vent
3 - Unit 1Turbine Building – NW Corner
4 - Unit 2 Turbine Building – SW Corner
5 - Unit 1 East Main Steam Enclosure
6 - Unit 2 East Main Steam Enclosure
7 - Unit 1 RWST
8 - Unit 2 RWST
9 - North Auxiliary Building Supply
10 - South Auxiliary Building Supply
11 - Unit 1 Containment Surface
12 - Unit 2 Containment Surface
13 - Unit 1 West Main Steam Enclosure
14 - Unit 2 West Main Steam Enclosure

Table 2.3-2: Onsite Release-Receptor Combination Parameters

Release-Receptor Pair	Release Location	Receptor Location (CR Intake)	Release Height (m)	Receptor Height (m)	Distance (m)	Direction (deg)	Building Area (m^2)
A	U1 Plant Vent	U1 Normal	49.5	15.3	44.4	79	1690
B	U1 Plant Vent	U1 Emergency	49.5	14.7	42.0	77	1690
C	U2 Plant Vent	U2 Normal	49.5	14.4	44.4	133	1690
D	U2 Plant Vent	U2 Emergency	49.5	14.7	42.0	135	1690
E	U2 Containment Closest Pt.	U2 Normal	14.4	14.4	25.8	133	1690
F	U2 Containment Closest Pt.	U2 Emergency	14.7	14.7	23.4	136	1690
G	U2 PORV/MSSV	U2 Normal	24.8	14.4	23.2	150	1690
H	U2 PORV/MSSV	U2 Emergency	24.8	14.7	21.4	155	1690
I	U2 Turbine Bldg	U2 Normal	14.4	14.4	9.8	286	0.01
J	U2 Turbine Bldg	U2 Emergency	14.7	14.7	12.5	286	0.01
K	U2 RWST	U2 Normal	12.9	14.4	68.6	161	0.01
L	U2 RWST	U2 Emergency	12.9	14.7	67.1	163	0.01
M	U2 Containment Surface (Diffuse)	U2 Normal	14.4	14.4	25.8	133	1690
N	U2 Containment Surface (Diffuse)	U2 Emergency	14.7	14.7	23.4	136	1690
O	U1 SJAE	U1 Normal	1.5	15.3	87.4	336	0.01
P	South Auxiliary Building Intake	U2 Normal	10.9	14.4	27.2	188	0.01

Table 2.3-3: Onsite Atmospheric Dispersion Factors

Release-Receptor Pair ID	Release Point	Receptor Point	0-2 hour *X/Q*	2-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
A	U1 Plant Vent	U1 Normal	2.03E-03	1.37E-03	4.89E-04	3.84E-04	2.62E-04
B	U1 Plant Vent	U1 Emergency	2.17E-03	1.42E-03	5.18E-04	3.99E-04	2.72E-04
C	U2 Plant Vent	U2 Normal	2.10E-03	1.48E-03	5.52E-04	3.98E-04	3.17E-04
D	U2 Plant Vent	U2 Emergency	2.28E-03	1.59E-03	5.95E-04	4.46E-04	3.49E-04
E	U2 Containment Closest Pt.	U2 Normal	1.02E-02	8.41E-03	2.74E-03	2.66E-03	2.34E-03
F	U2 Containment Closest Pt.	U2 Emergency	1.24E-02	1.02E-02	3.32E-03	3.24E-03	2.84E-03
G	U2 PORV/MSSV	U2 Normal	1.09E-02	8.61E-03	2.87E-03	2.78E-03	2.50E-03
H	U2 PORV/MSSV	U2 Emergency	1.26E-02	9.72E-03	3.26E-03	3.17E-03	2.80E-03
I	U2 Turbine Bldg	U2 Normal	4.57E-02	3.14E-02	1.27E-02	8.30E-03	6.73E-03
J	U2 Turbine Bldg	U2 Emergency	2.91E-02	2.02E-02	8.14E-03	5.34E-03	4.32E-03
K	U2 RWST	U2 Normal	1.74E-03	1.44E-03	5.24E-04	4.42E-04	3.86E-04
L	U2 RWST	U2 Emergency	1.81E-03	1.45E-03	5.43E-04	4.43E-04	3.86E-04
M	U2 Containment Surface (Diffuse)	U2 Normal	2.51E-03	1.94E-03	6.66E-04	6.44E-04	5.61E-04
N	U2 Containment Surface (Diffuse)	U2 Emergency	2.73E-03*	2.05E-03	7.04E-04	6.89E-04	5.95E-04
O	U1 SJAE	U1 Normal	8.50E-04				
P	South Auxiliary Building Intake	U2 Normal	7.91E-03	5.93E-03	2.12E-03	1.50E-03	1.01E-03

* 0-2 hour value is from Unit 1 containment to Unit 1 emergency intake

Table 2.3-4: Offsite Release-Receptor Distances

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 Containment Vent	N	542	3,149
	NNE	604	3,149
	NE	702	3,149
	ENE	823	3,158
	E	1,525	3,176
	ESE	1,292	3,201
	SE	982	3,219
	SSE	695	3,219
	S	683	3,219
	SSW	610	3,219
	SW	610	3,219
	WSW	610	3,219
	W	599	3,211
	WNW	575	3,185
	NW	554	3,166
	NNW	542	3,152
Unit 2 Containment Vent	N	609	3,218
	NNE	609	3,218
	NE	769	3,218
	ENE	946	3,218
	E	1,573	3,211
	ESE	1,267	3,185
	SE	790	3,166
	SSE	632	3,152
	S	613	3,149
	SSW	542	3,149
	SW	542	3,149
	WSW	548	3,158
	W	565	3,176
	WNW	589	3,201
	NW	609	3,219
	NNW	609	3,218

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 Turbine Building-NW Corner	N	434	3,043
	NNE	442	3,054
	NE	580	3,086
	ENE	770	3,141
	E	1,563	3,207
	ESE	1,413	3,275
	SE	965	3,329
	SSE	786	3,317
	S	666	3,281
	SSW	615	3,235
	SW	566	3,186
	WSW	527	3,142
	W	483	3,103
	WNW	449	3,061
	NW	434	3,044
	NNW	434	3,043
Unit 2 Turbine Building-SW Corner	N	591	3,211
	NNE	644	3,259
	NE	793	3,301
	ENE	990	3,333
	E	1,695	3,305
	ESE	1,048	3,241
	SE	674	3,172
	SSE	563	3,110
	S	519	3,065
	SSW	435	3,045
	SW	434	3,043
	WSW	434	3,043
	W	439	3,051
	WNW	465	3,080
	NW	506	3,128
	NNW	548	3,164

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 East Main Steam Enclosure	N	551	3,146
	NNE	638	3,143
	NE	695	3,143
	ENE	810	3,144
	E	1,523	3,154
	ESE	1,267	3,174
	SE	965	3,192
	SSE	685	3,195
	S	680	3,202
	SSW	617	3,212
	SW	617	3,222
	WSW	623	3,232
	W	626	3,238
	WNW	597	3,209
	NW	569	3,181
	NNW	551	3,161
Unit 2 East Main Steam Enclosure	N	617	3,216
	NNE	684	3,206
	NE	764	3,198
	ENE	933	3,193
	E	1,548	3,184
	ESE	1,242	3,163
	SE	798	3,148
	SSE	620	3,144
	S	610	3,144
	SSW	550	3,144
	SW	550	3,153
	WSW	560	3,172
	W	583	3,198
	WNW	612	3,227
	NW	627	3,236
	NNW	617	3,227

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 RWST	N	508	3,112
	NNE	603	3,112
	NE	667	3,113
	ENE	783	3,126
	E	1,503	3,154
	ESE	1,305	3,187
	SE	1,010	3,223
	SSE	728	3,237
	S	719	3,247
	SSW	641	3,251
	SW	633	3,242
	WSW	620	3,230
	W	592	3,207
	WNW	552	3,168
	NW	524	3,136
	NNW	508	3,118
Unit 2 RWST	N	637	3,247
	NNE	644	3,251
	NE	805	3,242
	ENE	978	3,230
	E	1,587	3,207
	ESE	1,256	3,168
	SE	763	3,136
	SSE	600	3,118
	S	577	3,113
	SSW	508	3,113
	SW	508	3,113
	WSW	515	3,127
	W	539	3,155
	WNW	572	3,188
	NW	613	3,223
	NNW	627	3,237

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
North Auxiliary Building Supply	N	525	3,134
	NNE	526	3,135
	NE	705	3,147
	ENE	816	3,170
	E	1,549	3,200
	ESE	1,327	3,234
	SE	898	3,247
	SSE	716	3,237
	S	694	3,223
	SSW	599	3,209
	SW	587	3,196
	WSW	577	3,187
	W	563	3,175
	WNW	540	3,152
	NW	527	3,137
	NNW	525	3,134
South Auxiliary Building Supply	N	592	3,202
	NNE	606	3,216
	NE	786	3,230
	ENE	908	3,242
	E	1,609	3,246
	ESE	1,299	3,213
	SE	808	3,181
	SSE	642	3,155
	S	611	3,139
	SSW	525	3,134
	SW	525	3,134
	WSW	525	3,134
	W	534	3,144
	WNW	553	3,165
	NW	575	3,184
	NNW	581	3,191

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 Containment Surface	N	523	3,130
	NNE	585	3,130
	NE	683	3,130
	ENE	804	3,139
	E	1,506	3,157
	ESE	1,273	3,182
	SE	963	3,200
	SSE	676	3,200
	S	664	3,200
	SSW	591	3,200
	SW	591	3,200
	WSW	591	3,200
	W	580	3,192
	WNW	556	3,166
	NW	535	3,147
	NNW	523	3,133
Unit 2 Containment Surface	N	590	3,199
	NNE	590	3,199
	NE	750	3,199
	ENE	927	3,199
	E	1,554	3,192
	ESE	1,248	3,166
	SE	771	3,147
	SSE	613	3,133
	S	594	3,130
	SSW	523	3,130
	SW	523	3,130
	WSW	529	3,139
	W	546	3,157
	WNW	570	3,182
	NW	590	3,200
	NNW	590	3,199

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 West Main Steam Enclosure	N	541	3,150
	NNE	541	3,150
	NE	715	3,158
	ENE	833	3,176
	E	1,549	3,200
	ESE	1,313	3,227
	SE	881	3,231
	SSE	700	3,221
	S	680	3,211
	SSW	592	3,202
	SW	586	3,195
	WSW	583	3,192
	W	577	3,186
	WNW	555	3,167
	NW	544	3,153
	NNW	541	3,150
Unit 2 West Main Steam Enclosure	N	589	3,198
	NNE	596	3,206
	NE	770	3,215
	ENE	891	3,226
	E	1,595	3,235
	ESE	1,294	3,210
	SE	812	3,185
	SSE	649	3,166
	S	623	3,153
	SSW	541	3,150
	SW	541	3,150
	WSW	542	3,151
	W	550	3,160
	WNW	566	3,178
	NW	583	3,192
	NNW	584	3,193

Table 2.3-5: Downwind Direction- Met Data Correlation

Direction	Met Data Set	
	EAB	LPZ
N	Shoreline	Shoreline
NNE	Shoreline	Primary
NE	Primary	Primary
ENE	Primary	Primary
E	Primary	Primary
ESE	Primary	Primary
SE	Primary	Primary
SSE	Shoreline	Primary
S	Shoreline	Primary
SSW	Shoreline	Shoreline
SW	Shoreline	Shoreline
WSW	Shoreline	Shoreline
W	Shoreline	Shoreline
WNW	Shoreline	Shoreline
NW	Shoreline	Shoreline
NNW	Shoreline	Shoreline

Table 2.3-6: Offsite Atmospheric Dispersion Factors

Release-Receptor Pair ID	Release Point	Receptor Point	0-2 hour *X/Q*	0-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
Q	U1 Plant Vent	EAB	5.69E-04				
R	U1 Plant Vent	LPZ	1.13E-04	5.28E-05	3.62E-05	1.64E-05	6.30E-06
S	U1 Containment Surface	EAB	6.04E-04				
T	U1 Containment Surface	LPZ	1.13E-04	5.32E-05	3.64E-05	1.66E-05	6.37E-06
U	U1 Turbine Bldg	EAB	8.62E-04				
V	U1 Turbine Bldg	LPZ	1.16E-04	5.45E-05	3.74E-05	1.74E-05	6.74E-06
W	U1 West Main Steam Enclosure	EAB	5.87E-04				

Release-Receptor Pair ID	Release Point	Receptor Point	0-2 hour X/Q	0-8 hour X/Q	8-24 hour X/Q	1-4 days X/Q	4-30 days X/Q
X	U1 Main Steam Enclosures	LPZ	1.13E-04	5.29E-05	3.63E-05	1.65E-05	6.36E-06
Y	U1 RWST	EAB	6.25E-04				
Z	U1 RWST	LPZ	1.14E-04	5.35E-05	3.67E-05	1.65E-05	6.36E-06
AA	North Auxiliary Building Supply	EAB	6.19E-04				
BB	North Auxiliary Building Supply	LPZ	1.13E-04	5.31E-05	3.64E-05	1.67E-05	6.42E-06

Table 2.3-7: Release-Receptor Pairs Application to the Event Analyses

Event	Normal Intake (1)	Emergency Intake (1)	EAB	LPZ
LOCA:				
- Containment Purge	C	D	Q	R
- Containment Leakage	M	N	S	T
- ESF Leakage	C	D	Q	R
- RWST Backleakage	K	L	Y	Z
FHA				
- Containment Release	E	F	S	T
- Auxiliary Bldg. Release	A	B	Q	R
MSLB:				
- Break Release	I	J	U	V
- Intact SG Release	G	H	U	V
SGTR	O, G (2)	H	U, W (3)	V, X (3)
Locked Rotor	G	H	W	X
Control Rod Ejection:				
- Containment Leakage	M	N	S	T
- Secondary Side Release	G	H	W	X
WGDT Rupture	P	n/a	AA	BB
VCT Rupture	P	n/a	AA	BB

(1) Control room makeup flow enters through the normal intake prior to realignment of the control room ventilation system and enters through the emergency intake after control room isolation. Unfiltered inleakage enters the control room envelope through the normal intake for the duration of the event.

(2) Prior to reactor trip, the release receptor pair is from the SJAE to the normal intake. The release point changes to the PORV/MSSV immediately after the trip, and the receptor point shifts to the emergency intake following control room isolation.

(3) Prior to reactor trip, the release is from the turbine building. The release point changes to the west main steam enclosure following the trip.

The breathing rates at EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8.

Table 2.3-8: Offsite Breathing Rates

Time (hours)	EAB/LPZ (m^3/sec)
0.0	3.5×10^{-4}
8.0	1.8×10^{-4}
24.0	2.3×10^{-4}
720.0	2.3×10^{-4}

2.4 *Direct Shine Dose*

In addition to the dose from contamination of the control room atmosphere by intake or infiltration, the total control room dose also requires consideration of direct shine dose contributions from control room filters, from the external radiation plume, and from radioactive material in the containment building. The filter shine dose is calculated by first determining the maximum activity loading on the control room ventilation system filters during the LOCA event. This is done by considering the control room ventilation maximum fan capacity flow rate along with filter efficiencies of 100%. The activities from the recirculation filter edit of the RADTRAD output files are then input into a MicroShield 8.03 model that reflects the geometry of the control room filter housing and the recirculation air handler unit position with respect to the control room. Credit is taken for shielding by structural materials and attenuation in air. An integrated 30-day dose is calculated for control room personnel.

The control room dose due to direct radiation streaming through the equipment hatch following a LOCA conservatively assumes that the control room receptor is positioned directly in front of the equipment hatch such that the exposure from containment is due to a direct line of sight through the entire area of the hatch. The analysis also under-estimates the total thickness of structural walls in the Auxiliary Building. The result is a conservative 30-day dose to an individual in the control room due to direct shine through the containment equipment hatch.

The dose contribution from the external cloud is assessed qualitatively using the guidelines of Reference [4.22] which states that 18 inches of concrete is generally adequate to attenuate the external DBA radiation to negligible levels. A review of site drawings revealed that while the minimum thickness of the control room walls is 18 inches, the control room is surrounded by 3 ft thick Auxiliary Building and 3'-6" Turbine Building walls. Similarly, the thickness of the concrete ceiling immediately above the control room is 18"; however, the thickness of the concrete roof on the elevation above the control room is 2'-6". As such, the shielding against external radiation sources well exceeds the amount identified as adequate in the guidance. Therefore, the additional shine dose contribution to control room personnel from the radioactive plume would be negligible.

The total LOCA shine dose is presented in Table 2.4-1. The filter shine dose following the LOCA event is conservatively applied to all other events.

Table 2.4-1: LOCA Direct Shine Dose

Source	Dose (rem)
Containment	0.246
Control Room Filters	0.139
External Cloud	Negligible
Total	0.385

3 Event Analyses

3.1 *Loss of Coolant Accident*

Control room and offsite doses are calculated for the LOCA event using the methodology outlined in Appendix A of Reference [4.1]. The dose contribution from the following four different radionuclide release pathways are determined separately and then combined to obtain the total dose for the event:

- Containment Purge
- Containment Leakage
- ESF Leakage to the Auxiliary Building
- ESF Leakage to the Refueling Water Storage Tank (RWST)

For all four cases, the control room ventilation system is automatically placed into the pressurization mode upon receipt of a safety injection signal. Parameters used in modeling the control room ventilation system are shown in Table 2.1-1.

3.1.1 Containment Purge

This containment purge release pathway represents releases through the Containment Purge Supply and Exhaust System prior to containment isolation. Since the purge system is isolated within 15 seconds following the initiation of the event, the release is secured well before the onset of the gap release at 30 seconds as defined in Table 4 of Reference [4.1]. Therefore, only those isotopes initially contained in the RCS fluid (Table 2.2-4) are available for release from containment, which are assumed to be instantaneously and homogeneously mixed throughout the containment atmosphere at the initiation of the event. The containment is modeled as a single compartment without credit for isotope removal by sprays or deposition. Radionuclides are released from containment directly to the environment without mitigation until the containment purge system is isolated. In addition, there is no radionuclide reduction by the containment purge ventilation system, which exhausts to the plant vent.

3.1.2 Containment Leakage

For the containment leakage case, 100% of the core becomes damaged and a phased release of the core fission product inventory (Table 2.2-2) occurs. The fraction of the nuclides in the core that become deposited into containment and the timing of this release are shown in Table 3.1-1 and Table 3.1-2.

Table 3.1-1: Core Inventory Fraction Release into Containment

Group Number	Group Name	Elements	Gap Release Phase	Early In-Vessel Phase
1	Noble Gases	Xe, Kr	0.05	0.95
2	Halogens	I, Br	0.05	0.35
3	Cesium	Cs, Rb	0.05	0.25
4	Tellurium	Te, Sb, Se	0.0	0.05
5	Strontium	Sr	0.0	0.02
6	Barium	Ba	0.0	0.02
7	Ruthenium	Ru, Rh, Pd, Mo, Tc, Co	0.0	0.0025
8	Cerium	Ce, Pu, Np	0.0	0.0005
9	Lanthanum	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0.0	0.0002

Table 3.1-2: LOCA Release Phase Timing

Phase	Onset	Duration
Gap Release	30 sec	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr

The released nuclides are assumed to be instantaneously and homogeneously mixed throughout the containment. The D. C. Cook containment building is modeled as seven separate regions. The entire containment building is served by the safety related Containment Ventilation (CEQ) System. The CEQ System does not have any filtration capability; however, it does provide some additional level of mixing between the regions. Three of the regions are capable of being sprayed by the Containment Spray (CTS) System. Spray induced mixing is also credited between adjacent sprayed and unsprayed regions based on two turnovers of the unsprayed volume per hour. The CTS System is assumed to be secured after 24 hours.

Iodine is released into the containment with a chemical composition of 95% particulate, 4.85% elemental, and 0.15% organic. The containment sump pH is maintained greater than 7.0 following the onset of containment sprays; therefore, re-evolution of particulate iodine into elemental iodine is not considered. Spray removal of elemental and aerosol iodine is credited using the guidance of Reference [4.23]. Removal of elemental iodine in each of the sprayed regions is terminated when the elemental decontamination factor in the region reaches a value of 200. Similarly, the removal rate of the aerosol iodine is reduced by a factor of 10 when the aerosol decontamination factor reaches 50. Natural deposition of only the aerosol iodine is considered in the analysis, and deposition is credited only in unsprayed regions and in sprayed regions after the CTS system has been secured.

Unfiltered leakage from the containment to the environment is assumed to occur uniformly from all seven regions at an initial rate of 0.18%/day. This leakage rate is reduced by 50% to 0.09%/day after 24 hours. The release from the containment is based upon an atmospheric dispersion factor assuming a diffuse source from the containment surface.

3.1.3 ESF Leakage into the Auxiliary Building

ESF leakage outside of containment results from operation of ECCS systems which take suction from the containment sump and allow system fluid to be released into the Auxiliary Building through pump seals, valve packing glands, and flanged connections. For this case, a portion of the core source term from Table 2.2-2 is deposited into liquid in the containment sump according to the release fraction and timing shown in Table 3.1-1 and Table 3.1-2. With the exception of noble gases, the fission products released from the fuel are assumed to instantaneously and homogeneously mix in the containment sump water. Leakage from the ECCS systems begins at the onset of switchover to recirculation, and the leak rate into the building is taken as two times the allowable limit established by the leak rate monitoring program. Once the sump fluid exits the system, particulate nuclides are assumed to be retained in the liquid phase, which limits the release to iodine isotopes only. Ten percent of the iodines in the sump fluid are then assumed to become airborne and are released directly to the environment. No credit is taken for holdup or dilution in the Auxiliary Building, or for filtration removal by the ESF Ventilation system. All releases from the Auxiliary Building occur from the plant vent.

3.1.4 ESF Leakage into the RWST

The evaluation of ESF leakage through valves that isolate interfacing systems from the RWST is performed in a separate case. The sump activity for this case is identical to that applied in the case of ESF leakage to the Auxiliary Building described in Section 3.1.3. For this case, flow from the sump into the RWST is assumed to begin immediately upon switchover to recirculation at a rate of 1 gpm (two times 0.5 gpm). The modeling of the release of volatile iodine from the tank to the atmosphere is based upon the guidance of NUREG/CR-5950 (Reference [4.25]). Based upon sump pH controls, the iodine in the sump is considered to be nonvolatile. However, when introduced into the acidic solution of the RWST, a portion of the particulate iodine is converted to elemental iodine. The methodology of NUREG/CR-5950 accounts for two iodine transport/release mechanisms. The first is the fraction of the total iodine in the tank that is released in the form elemental iodine, and the second is the partitioning of elemental iodine between the liquid and vapor phases of the tank.

The fraction of the total iodine that becomes elemental is a function of both the RWST pH and the total iodine concentration in the tank. The analysis determines the time-dependent RWST pH profile as the sump fluid, with a constant pH of 7.0, mixes with the remaining inventory in the tank following switchover to recirculation. Similarly, the total RWST iodine concentration is calculated as sump iodine is transported into the tank from the sump. Calculation of the time-dependent iodine concentration in the RWST liquid for purposes of determining the elemental iodine release fraction conservatively neglects the reduction in concentration due to the release of iodine into the vapor phase. These two parameters combine to produce the elemental iodine fraction, which increases from a value of 0.0 at the beginning of the event to a maximum of 0.1914.

The ratio of the elemental iodine concentrations between the liquid and vapor phases of the tank is determined by a partition coefficient that is a function of the RWST liquid temperature. The analysis calculates a conservatively high RWST temperature profile using GOTHIC by introducing hot sump fluid into the tank without credit for heat removal in the piping between the sump and the tank or heat losses through the tank walls. This results in a time-dependent partition coefficient that decreases from 45.41 at the beginning of the event to 31.92 after 30 days. The elemental iodine fraction and partition coefficient are applied to the leakage flow rate from the sump to the RWST to obtain an adjusted elemental iodine release rate from the tank. A similar approach is taken with the organic iodine, using a release fraction of 0.0015 taken from Position 2 of Appendix A to Reference [4.1] and assuming a conservative partition coefficient of 1.0. The release location for this event is from the RWST vent.

Values of key inputs and assumptions important to the LOCA analysis are provided in Table 3.1-3. The dose consequences for this event are presented in Table 3.1-10.

Table 3.1-3: LOCA Inputs and Assumptions

Input/Assumption	Value
Containment Purge	
Source Term	Initial RCS Activity (Table 2.2-4)
Iodine Chemical Form	95% aerosol, 4.85% elemental, 0.15% organic
Containment Volume	1,066,352 ft^3 (minimum)
Containment Purge Flow Rate	36,300 cfm
Containment Purge Isolation time	15 seconds
Containment Purge Filtration	0%
Removal by Wall Deposition	None
Removal by Sprays	None
Containment Leakage	
Source Term	Core Inventory (Table 2.2-2)
Iodine Chemical Form	95% aerosol, 4.85% elemental, 0.15% organic
Containment Sump pH	>7.0
Compartment Volumes (max) Upper Containment (Sprayed) Lower Containment (Sprayed) Fan Rooms (Sprayed) Upper Containment (Unsprayed) Ice Condenser (Unsprayed) Lower Containment (Unsprayed) Dead-End (Unsprayed)	 621,968 ft^3 103,770 ft^3 48,913 ft^3 122,600 ft^3 105,577 ft^3 66,188 ft^3 18,663 ft^3

Input/Assumption	Value
Containment Ventilation Start Time	300 seconds
Containment Ventilation Flow Rate Fan Rooms to Lower Containment (Unsprayed) Fan Rooms to Lower Containment (Sprayed) Lower Containment (Unsprayed) to Dead –End Dead-End to Fan Rooms Lower Containment (Unsprayed) to Fan Rooms Lower Containment (Unsprayed) to Ice Condenser Lower Containment (Sprayed) to Ice Condenser Ice Condenser to Upper Containment (Sprayed) Ice Condenser to Upper Containment (Unsprayed) Upper Containment (Sprayed) to Fan Rooms Upper Containment (Unsprayed) to Fan Rooms Lower Containment – Sprayed to/from Unsprayed Upper Containment – Sprayed to/from Unsprayed	 14,580.5 cfm 22,859.5 cfm 90 cfm 90 cfm 1,350 cfm 13,140.5 cfm 22,859.5 cfm 30,072.3 cfm 5,927.7 cfm 30,072.3 cfm 5,927.7 cfm 2206.3 cfm (spray induced circulation) 4086.7 cfm (spray induced circulation)
Sprayed/Unsprayed Volume Induced Mixing Flow Rate	2 Turnovers of Unsprayed Compartment/hour
Containment Spray Start Time	300 seconds
Containment Spray Stop Time	0.319-0.426 hours and after 24 hours
Containment Spray Flow Rate Upper Containment Lower Containment Fan Rooms	 1466 gpm 660 gpm 201 gpm
Containment Spray Drop Fall Height Upper Containment Lower Containment Fan Rooms	 58.6 ft 28.5 ft 20.1 ft
Containment Spray Mean Drop Diameter Upper Containment Lower Containment Fan Rooms	 609 microns 671 microns 671 microns
Elemental Iodine Spray Removal Coefficient	20 hr^{-1}, with a total decontamination factor of 200
Time that Total Elemental DF reaches 200	2 hours

Input/Assumption	Value
Aerosol Spray Removal Coefficient Upper Containment Lower Containment Fan Rooms	 5.06 hr^{-1} 6.65 hr^{-1} 3.03 hr^{-1}
Time that Total Aerosol DF reaches 50	2.32 hours
Organic Iodine Spray Removal	None
Natural Deposition	Elemental, Organic Iodine – None Aerosols - 0.1 hr^{-1} in unsprayed regions only
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.18 %/day 0.09 %/day
Containment Leakage Filtration	0%
ESF Leakage to the Auxiliary Building	
Source Term	Core Inventory (Table 2.2-2)
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic
Containment Sump Volume	50,955 ft^3
ECCS Recirculation Start Time	1388.4 seconds
ESF Leakage Flow Rate	0.2 gpm (two times the allowable value)
ESF Leakage Flashing Fraction	10%
Auxiliary Building Ventilation Filtration	0%
ESF Leakage to the RWST	
Source Term	Core Inventory (Table 2.2-2)
Containment Sump Volume	50,955 ft^3
ECCS Recirculation Start Time	1388.4 seconds
ESF Leakage Flow Rate	1.0 gpm (two times the allowable value)
Total iodine mass released into the sump	12,035.5 grams
Sump iodine concentration	6.573×10^{-05} g-atom/liter
Sump pH	7.0
Total RWST Volume	420,000 gallons
Initial RWST Liquid Volume	53,637.5 gallons (minimum at time of switchover)
RWST Liquid Iodine Concentration	0.0 - 2.931×10^{-05} g-atom/liter (Table 3.1-4)

Input/Assumption	Value
RWST pH	4.479 – 4.734 (Table 3.1-5)
Elemental Iodine Release Fraction	0.0 - 0.1914 (Table 3.1-6)
Organic Iodine Fraction	0.0015
RWST Liquid Temperature	100 – 118.5 °F (Table 3.1-7)
RWST Liquid/Vapor Iodine Partition Coefficient Elemental Organic	 31.92 - 45.41 (Table 3.1-8) 1.0
Adjusted RWST Iodine Release Rate	Table 3.1-9
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-6 and Table 2.3-7 Table 2.3-3 and Table 2.3-7
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8. 3.5×10^{-4} m^3/sec
Control Room Model Inputs	Table 2.1-1
Control Room Isolation Time	70 seconds (Safety Injection)
Control Room Occupancy Factor	Table 2.1-1

Table 3.1-4: RWST Liquid Iodine Concentration

Time (hr)	Iodine Concentration (g-atom/liter)
0.0	0.000E+00
0.386	0.000E+00
0.50	8.381E-09
1.0	4.512E-08
5.0	3.375E-07
10.0	6.994E-07
15.0	1.057E-06
25.0	1.761E-06
50.0	3.456E-06
75.0	5.064E-06
100.0	6.590E-06
125.0	8.042E-06
150.0	9.424E-06
200.0	1.200E-05
250.0	1.435E-05
300.0	1.650E-05
350.0	1.848E-05
400.0	2.031E-05
450.0	2.200E-05
500.0	2.357E-05
550.0	2.503E-05
600.0	2.639E-05
650.0	2.766E-05
700.0	2.886E-05
720.0	2.931E-05

Table 3.1-5: RWST pH

Time (hr)	pH
0.0	4.479
0.386	4.479
0.50	4.479
1.0	4.479
5.0	4.481
10.0	4.484
15.0	4.486
25.0	4.491
50.0	4.502
75.0	4.514
100.0	4.525
125.0	4.535
150.0	4.546
200.0	4.566
250.0	4.586
300.0	4.604
350.0	4.622
400.0	4.639
450.0	4.655
500.0	4.671
550.0	4.686
600.0	4.701
650.0	4.715
700.0	4.729
720.0	4.734

Table 3.1-6: Elemental Iodine Release Fraction

Time (hr)	I_2 Fraction
0.0	0.0000
0.386	0.0000
0.50	0.0002
1.0	0.0011
5.0	0.0080
10.0	0.0161
15.0	0.0238
25.0	0.0379
50.0	0.0673
75.0	0.0902
100.0	0.1085
125.0	0.1234
150.0	0.1356
200.0	0.1541
250.0	0.1670
300.0	0.1761
350.0	0.1824
400.0	0.1866
450.0	0.1893
500.0	0.1908
550.0	0.1914
600.0	0.1914
650.0	0.1907
700.0	0.1896
720.0	0.1890

Table 3.1-7: RWST Temperature Profile

Time (hr)	Temperature °F
0.0	100.0
0.386	100.0
0.50	100.0
1.0	100.1
5.0	100.3
10.0	100.5
15.0	100.8
25.0	101.3
50.0	102.4
75.0	103.5
100.0	104.5
125.0	105.5
150.0	106.4
200.0	108.1
250.0	109.7
300.0	111.1
350.0	112.2
400.0	113.3
450.0	114.3
500.0	115.2
550.0	116.0
600.0	116.8
650.0	117.5
700.0	118.2
720.0	118.5

Table 3.1-8: RWST I_2 Partition Coefficient

Time (hr)	Partition Coefficient
0.0	45.41
0.386	45.41
0.50	45.41
1.0	45.33
5.0	45.15
10.0	44.98
15.0	44.73
25.0	44.30
50.0	43.38
75.0	42.48
100.0	41.68
125.0	40.89
150.0	40.20
200.0	38.92
250.0	37.75
300.0	36.75
350.0	35.99
400.0	35.24
450.0	34.58
500.0	33.99
550.0	33.48
600.0	32.97
650.0	32.53
700.0	32.10
720.0	31.92

Table 3.1-9: Adjusted RWST Iodine Release Rate

Time (hr)	Release Rate (cfm)
0.0	0.0
0.386	6.790E-07
10.0	2.969E-06
25.0	1.300E-05
75.0	3.318E-05
125.0	6.398E-05
200.0	1.118E-04
300.0	1.800E-04
450.0	2.560E-04
600.0	3.167E-04
720.0	3.167E-04

Table 3.1-10: LOCA TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Containment Purge	1.0601E+00	2.1053E-01	7.2674E-01
Containment Leakage	1.7972E+01	4.5696E+00	1.7181E+00
ESF Leakage	2.4192E+00	3.4498E+00	1.6848E+00
RWST Backleakage	2.1091E-02	6.2971E-02	3.9434E-02
Control Room Shine			0.385
Total	**21.48**	**8.30**	**4.56**
Acceptance Limit	25	25	5

3.2 *Fuel Handling Accident*

The Fuel Handling Accident is evaluated as a drop of a single fuel assembly in which 100% of the rods in the dropped assembly are assumed to fail. The analysis considers both a drop in the containment building without established containment integrity, and a drop in the Auxiliary Building with the Fuel Handling Area Exhaust Ventilation (FHAEV) in service. The source term for this event is described in Section 2.2.2; however, the nuclides available for release are limited to those isotopes present in the fuel rod gap listed in Table 2.2-6. As discussed in Section 2.2.5, the gap inventories shown in Table 2.2-6 are doubled to account for the presence of high burnup fuel rods in the dropped assembly.

The analysis of the Fuel Handling Accident in both locations is modeled as a fuel assembly drop which occurs 120 hours after reactor shutdown and results in the activity which escapes from the pools being released to the environment over a 2-hour period. The water in the pool is credited with a decontamination factor of 285 for elemental iodine and 1.0 for organic iodine as discussed in Item 8 of Reference [4.2]. The pool water is assumed to retain 100% of the alkali metals. It is also assumed that the pool water will have no impact on the noble gases. No credit is taken for mixing or holdup in either the Auxiliary Building or the containment. However, iodine removal by the FHAEV system filters is modeled. The release location for the FHA in containment for the control room dose is assumed to be a point on the external containment surface closest to the control room intakes. For the drop in the Auxiliary Building, the FHAEV system discharges to the plant vent. The control room is assumed to be manually placed into the pressurization mode 20 minutes after the start of the event by the plant operators. Control room ventilation parameters are listed in Table 2.1-1.

Major inputs and assumptions important to this event are provided in Table 3.2-1. The dose consequences are presented in Table 3.2-2 and Table 3.2-3. Note that the direct shine dose contribution from the control room filters conservatively reflects values from the LOCA event.

Table 3.2-1: Fuel Handling Inputs and Assumptions

Input/Assumption	Value
Source Term	FHA Inventory (Table 2.2-3)
Iodine Chemical Form	0% aerosol, 99.85% elemental, 0.15% organic
Number of Fuel Assemblies Damaged	1
Percentage of Fuel Rods Failed	100%
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	150
High burnup multiplier applied to gap fractions	2.0
Water Level Above Damaged Fuel	23 feet
Pool Decontamination Factors	Elemental – 285 Organic – 1.0
Delay Before Fuel Movement	120 hours
Containment Release Filtration	0%

Input/Assumption	Value
Fuel Handling Area Exhaust Ventilation Filtration	Aerosol - 98.01% Elemental - 89.1% Organic - 89.1%
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-6 and Table 2.3-7 Table 2.3-3 and Table 2.3-7
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	20 minutes (Manual)
Control Room Occupancy Factor	Table 2.1-1

Table 3.2-2: Fuel Handling Accident – Containment Release TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Containment Release	3.5676E+00	6.6745E-01	4.3492E+00
Control Room Shine			0.139
Total	**3.57**	**0.67**	**4.49**
Acceptance Limit	6.3	6.3	5

Table 3.2-3: Fuel Handling Accident – Auxiliary Building Release TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Auxiliary Building Release	6.7137E-01	1.3343E-01	1.3039E-01
Control Room Shine			0.139
Total	**0.68**	**0.14**	**0.27**
Acceptance Limit	6.3	6.3	5

3.3 *Main Steam Line Break*

This event consists of a break in one main steam line outside of containment in which the faulted steam generator (SG) completely depressurizes and instantly releases the initial contents of the steam generator secondary side to the environment. The plant cooldown continues by dumping steam with the intact steam generators. In addition to the release of nuclides that are initially present in the steam generator secondary side, leakage of primary coolant into the steam generator secondary side occurs at a rate equal to the proposed Tech. Spec. program limit of 0.25 gpm/SG.

This event does not result in fuel damage. Consequently, two iodine spike cases are considered. In the first (pre-accident spike) case, a reactor transient is assumed to occur prior to the MSLB in which the primary coolant iodine concentration has increased to the maximum Tech. Spec. value of 60 μCi/gm. In the second (concurrent spike) case, the iodine release rate into the primary coolant increases to a value that is 500 times greater than the 'normal' release rate that corresponds to the Tech. Spec. specific iodine limit of 1 μCi/gm. This concurrent iodine spike is assumed to have a duration of 8 hours. Inputs used in the development of the concurrent iodine spike appearance rate are shown in Table 3.3-1 and the 8-hour total iodine activity released into the reactor coolant is provided in Table 3.3-2. In both cases, the remaining non-iodine isotopes in the RCS listed in Table 2.2-4 are also available for release.

Leakage from the RCS into all of the steam generators, and steam release from the intact steam generators, continues until the RCS is cooled to 212 °F after 24 hours. During this period, all of the noble gases and all of the nuclides which leak into the faulted steam generator are released directly to the environment without mitigation. Leakage into the intact steam generators mixes with the bulk fluid where a portion of the activity is released based upon the steaming rate and a partition coefficient. A partition coefficient of 100 is applied to the iodine nuclides, and the particulate release is limited by the moisture carryover. It is recognized that early in the transient, the water level in the intact steam generator secondary may be below the top of the tube bundle and the bulk water partitioning may not apply. In this case, a flashing fraction is calculated based upon the thermodynamic conditions in the reactor and secondary coolant. The portion of the primary-to-secondary leakage which flashes to vapor is assumed to be released directly to the environment without mixing. The iodine and particulate partition coefficients are applied to the unflashed portion. The tube bundles in the intact steam generators are assumed to be fully covered after 40 minutes.

The release locations from the faulted steam generator are selected to maximize the control room and offsite doses without regard to the location of the break with respect to the main steam isolation valves (MSIV) and

without credit for MSIV closure. Releases from the intact steam generators occur from the PORVs/MSSVs. The control room is automatically realigned into the pressurization mode (Table 2.1-1) upon receipt of a safety injection signal.

Parameters important to the analysis of the MSLB event are shown in Table 3.3-3, and the analysis results are provided in Table 3.3-4 and Table 3.3-5.

Table 3.3-1: Main Steam Line Break Iodine Appearance Rate Inputs and Assumptions

Input/Assumption	Value
Letdown Flow Rate	132 gpm
Identified RCS Leakage	10 gpm
Unidentified RCS Leakage	1 gpm
RCS Mass	607,290.6 lbm (maximum)
I-131 Decay Constant	0.000060 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013176 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

Table 3.3-2: Main Steam Line Break 500x Iodine Appearance

Isotope	Appearance Rate (Ci/min)	8-Hour Production (Ci)
I-131	223.45	107,256
I-132	615.35	295,368
I-133	354.95	170,376
I-134	256.40	123,096
I-135	272.95	131,016

Table 3.3-3: Main Steam Line Break Inputs and Assumptions

Input/Assumption	Value
Source Term	Initial RCS Activity (Table 2.2-4)
Maximum Pre-Accident Iodine Spike Concentration	60 µCi/gm Dose Equivalent I-131
Concurrent Iodine Spike Appearance Rate	500x Equilibrium (Table 3.3-2)
Initial Steam Generator Iodine Source Term	0.1 µCi/gm Dose Equivalent I-131
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic
Percentage of Fuel Rods Failed	0%
RCS Mass	466,141.5 lbm (minimum)
Steam Generator Secondary Liquid Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Intact Steam Generator Steam Release	0 - 2 hours: 456,000 lbm 2 - 8 hours: 1,186,000 lbm 8 - 24 hours: 1,347,000 lbm
Primary-Secondary Leak Rate	0.25 gpm to each steam generator
Density Used for Leakage Volume-to-Mass Conversion	62.3 lbm/ft^3
Duration of Intact SG Tube Uncovery After Reactor Trip	40 minutes
Tube Leakage Flashing Fraction During Uncovery	0- 400 seconds: 8% 400-900 seconds: 6% 900-1700 seconds: 5.5% 1700 seconds-40 min: 4%
Time to Cool RCS to 212 °F	24 hours
Intact Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0
Intact Steam Generator Moisture Carryover Fraction	0.2% (Particulate Partition Coefficient = 500)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-6 and Table 2.3-7 Table 2.3-3 and Table 2.3-7
Dose Conversion Factors Breathing Rates EAB/LPZ	FGR 11 & FGR 12 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8.

Input/Assumption	Value
Control Room	3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	70 seconds (Safety Injection)
Control Room Occupancy Factor	Table 2.1-1

Table 3.3-4: Main Steam Line Break Pre-Accident Spike TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas	4.2923E-04	2.0284E-04	1.7912E-03
Pre-Accident Iodine Spike	1.6032E-01	7.8786E-02	4.7971E-01
Initial SG Secondary Iodine	8.0183E-02	1.1235E-02	7.5755E-01
Control Room Shine			0.139
Total	**0.25**	**0.10**	**1.38**
Acceptance Limit	25	25	5

Table 3.3-5: Main Steam Line Break Concurrent Spike TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas	4.2923E-04	2.0284E-04	1.7912E-03
Iodine Release	6.7612E-01	2.3694E-01	1.8792E+00
RCS Activity Release	8.1207E-02	3.9657E-02	2.0951E-01
Initial SG Secondary Iodine	8.0183E-02	1.1235E-02	7.5755E-01
Control Room Shine			0.139
Total	**0.84**	**0.29**	**2.99**
Acceptance Limit	2.5	2.5	5

3.4 *Steam Generator Tube Rupture*

The SGTR event represents an instantaneous rupture of a steam generator tube that releases primary coolant into the lower pressure secondary system. In addition to the break flow rate, primary-to-secondary leakage occurs at a rate equal to the proposed Tech. Spec. program limit of 0.25 gpm/SG to the ruptured and to each of the intact steam generators. All leakage flow into the ruptured steam generator is secured after 30 minutes. Leakage into the intact steam generators continues until the RCS is cooled to 212 °F after 24 hours. A portion of the break and leakage flow to the ruptured steam generator flashes to vapor based upon the thermodynamic conditions in the reactor and secondary coolant. The portion of the primary coolant that does flash in the steam generator secondary is released directly to the environment without mitigation. Although non-iodine particulates are not volatile and unlikely to be transported directly to the environment, the RCS fluid flashing fractions are conservatively applied to the particulate nuclides, which contributes significantly to the total dose. The unflashed break and leakage flow mixes with the bulk water in the steam generator where the activity is released based upon the steaming rate and a partition coefficient. A steam generator partition coefficient of 100 is applied to the iodine nuclides, and the particulate release is limited by the moisture carryover. Prior to the reactor trip, an additional condenser partition coefficient of 100 is applied to the released activity. This same approach is applied to the primary-to-secondary leakage into the intact steam generators at the start of the event when the SG tube bundles are assumed to become uncovered following the reactor trip. After 40 minutes, flashing of the leakage flow is no longer applicable because the tube bundles have become fully submerged.

This event results in no fuel damage, and as such, two iodine spike cases are considered. In the pre-accident iodine spike case, a reactor transient is assumed to occur prior to the event in which the primary coolant iodine concentration has increased to the maximum Tech. Spec. value of 60 μCi/gm. In the concurrent iodine spike case, the iodine release rate into the primary coolant increases to a value that is 335 times greater than the 'normal' release rate that corresponds to the Tech. Spec. specific iodine limit of 1 μCi/gm. This concurrent iodine spike is assumed to have a duration of 8 hours. Inputs used in the development of the concurrent iodine spike appearance rate are shown in Table 3.4-1, and the 8-hour total iodine activity released into the reactor coolant is provided in Table 3.4-2. In both cases, the remaining non-iodine isotopes in the RCS listed in Table 2.2-4 are also available for release. In addition to the activity transported into the steam generators from the primary coolant, the analysis considers the release of the iodine activity initially present in the steam generator secondary inventory.

Prior to the reactor trip, the activity is assumed to be released from the Steam Jet Air Ejector in the Turbine Building. Following the trip, the release location shifts to the PORVs/MSSVs. The control room is automatically realigned into the pressurization mode (Table 2.1-1) upon receipt of a safety injection signal.

Key inputs and assumptions applied in the analysis of the SGTR event are shown in Table 3.4-3, and the analysis results are provided in Table 3.4-4 and Table 3.4-5.

Table 3.4-1: SGTR Iodine Appearance Rate Inputs and Assumptions

Input/Assumption	Value
Letdown Flow Rate	132 gpm
Identified RCS Leakage	10 gpm
Unidentified RCS Leakage	1 gpm
RCS Mass	607,290.6 lbm (maximum)
I-131 Decay Constant	0.000060 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013176 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

Table 3.4-2: SGTR 335x Iodine Appearance

Isotope	Appearance Rate (Ci/min)	8-Hour Production (Ci)
I-131	149.71	71,861
I-132	412.28	197,894
I-133	237.82	114,154
I-134	171.79	82,459
I-135	182.88	87,782

Table 3.4-3: Steam Generator Tube Rupture Inputs and Assumptions

Input/Assumption	Value
Source Term	Initial RCS Activity (Table 2.2-4)
Maximum Pre-Accident Iodine Spike Concentration	60 μCi/gm Dose Equivalent I-131
Concurrent Iodine Spike Appearance Rate	335x Equilibrium (Table 3.4-2)
Initial Steam Generator Iodine Source Term	0.1 μCi/gm Dose Equivalent I-131
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic
Percentage of Fuel Rods Failed	0%
RCS Mass	466,141.5 lbm (minimum)
Steam Generator Secondary Liquid Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Intact Steam Generator Steam Release	0 - 30 min. - 198,515 lbm 30 min. - 2 hours: 314,432 lbm 2 - 8 hours: 1,367,475 lbm 8 - 24 hours: 1,347,000 lbm
Ruptured Steam Generator Steam Release	0 - 30 min. – 66,171 lbm
Pre-Trip Total Steam Flow Rate Through Condenser	17,153,800 lbm/hr
Time of Reactor Trip	101 seconds
Primary-Secondary Leak Rate	0.25 gpm to each steam generator
Density Used for Leakage Volume-to-Mass Conversion	62.3 lbm/ft^3
Ruptured Tube Break Flow	146,704 lbm
Duration of Ruptured Tube Break Flow	30 minutes
Break Flow Flashing Fraction	Pre- Trip 0-100 seconds: 19% Post-Trip: 100- 500 seconds: 8% 500-1000 seconds: 6% 1000-1800 seconds: 5.5%
Duration of Intact SG Tube Uncovery After Reactor Trip	40 minutes
Intact Tube Leakage Flashing Fraction During Uncovery	0-100 seconds: 19% 100- 500 seconds: 8% 500-1000 seconds: 6% 1000-1800 seconds: 5.5% 1800 seconds-40 min: 4%

Input/Assumption	Value
Time to Cool RCS to 212 °F	24 hours
Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0
Condenser Partition Coefficient	100
Steam Generator Moisture Carryover Fraction	0.2% (Particulate Partition Coefficient = 500)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-6 and Table 2.3-7 Table 2.3-3 and Table 2.3-7
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	394.74 seconds (Safety Injection)
Control Room Occupancy Factor	Table 2.1-1

Table 3.4-4: Steam Generator Tube Rupture Pre-Accident Spike TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas	4.1808E-02	7.8965E-03	2.3863E-02
Pre-Accident Iodine Spike	4.1154E+00	7.4309E-01	3.7895E+00
Initial SG Secondary Iodine	1.8818E-03	8.3936E-04	2.6022E-03
Control Room Shine			0.139
Total	**4.16**	**0.76**	**3.96**
Acceptance Limit	25	25	5

Table 3.4-5: Steam Generator Tube Rupture Concurrent Spike TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas	4.1808E-02	7.8965E-03	2.3863E-02
Iodine Release	4.0639E-01	8.8981E-02	2.4586E-01
RCS Activity Release	2.0463E+00	3.6688E-01	1.8432E+00
Initial SG Secondary Iodine	1.8818E-03	8.3936E-04	2.6022E-03
Control Room Shine			0.139
Total	**2.50***	**0.47**	**2.26**
Acceptance Limit	2.5	2.5	5

* Calculated value is 2.497 rem which is rounded up to 2.50.

3.5 *Locked Rotor*

The Locked Rotor dose analysis is defined by the 11% of the fuel rods which become damaged by the event. Radionuclides released from the fuel are instantaneously and homogeneously distributed throughout the primary coolant. Noble gases are released directly to the environment, and the remaining isotopes are transported to the steam generators at a rate of 1 gpm. The core source term from Table 2.2-2 is applicable to this event, and the fraction of these activities available for release into the coolant are based upon the gap inventory fractions shown in Table 2.2-6 and the assembly radial peaking factor of 1.65. To account for fuel rods contained in two of the fuel assemblies which exceed the burnup limits of Footnote 11 of Reference [4.1], the gap inventory of all of the rods in these two assemblies are assumed to be twice those listed in Table 2.2-6 as discussed in Section 2.2.5. As such, with 193 fuel assemblies in the core, the effective core-wide multiplier on the gap inventory fractions is 1 + (2/193) = 1.0104. Since the fuel failure fraction is applied to the entire core source term, 11% of the rods in both the standard and high burnup assemblies are assumed to fail, and the assembly peaking factor is conservatively applied to all of the failed rods in the core.

During the first 40 minutes of the event, the water level on the secondary side of the steam generators is assumed to be below the top of the tube bundles. During this time, a portion of the primary-to-secondary leakage flashes to vapor based upon the thermodynamic conditions of the reactor and secondary coolant. Nuclides contained in the flashed tube leakage are released to the environment without mitigation. The unflashed leakage mixes with the bulk water in the steam generators and is released as a function of the steaming rate and the partition coefficients. After 40 minutes, all of the leakage is treated as unflashed, which continues until 24 hours when the RCS temperature is cooled to 212 °F.

Since the quantity of the fission products released from the failed fuel dominates the RCS activity during the event, the initial nuclide concentration in the RCS prior to the event is not considered. However, the analysis does include the dose contribution from the release of iodine initially present in the steam generator secondary side. All releases occur from the PORVs/MSSVs, which are located in the Main Steam Enclosures. For this

event, the control room ventilation system remains in the normal alignment without filtration or recirculation until manually placed into the pressurization mode after 20 minutes.

Major inputs and assumptions applicable to the Locked Rotor event are listed in Table 3.5-1. The analysis results are presented in Table 3.5-2.

Table 3.5-1: Locked Rotor Inputs and Assumptions

Input/Assumption	Value
Source Term	Core Inventory (Table 2.2-2)
Fuel Rod Gap Fractions	I-131 - 0.08 Kr-85 - 0.10 Other Noble Gases - 0.05 Other Halogens - 0.05 Alkali Metals - 0.12
Percentage of Fuel Rods Failed	11%
Fuel Rod Peaking Factor	1.65
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	150 rods in two assemblies
High burnup multiplier applied to gap fractions	1.0104
Initial Steam Generator Iodine Source Term	0.1 μCi/gm Dose Equivalent I-131
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic
RCS Mass	466,141.5 lbm (minimum)
Steam Generator Secondary Liquid Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Primary-Secondary Leak Rate	1.0 gpm to all steam generators
Density Used for Leakage Volume-to-Mass Conversion	62.3 lbm/ft^3
Secondary Steam Release	0 - 2 hours: 460,000 lbm 2 - 8 hours: 1,256,000 lbm 8 - 24 hours: 1,347,000 lbm
Time to Cool RCS to 212 °F	24 hours
Duration of SG Tube Uncovery Following Reactor Trip	40 minutes
Intact Tube Leakage Flashing Fraction During Uncovery	0- 400 seconds: 8% 400-900 seconds: 6% 900-1700 seconds: 5.5% 1700 seconds-40 min: 4%
Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0

Input/Assumption	Value
Steam Generator Moisture Carryover Fraction	0.2% (Particulate Partition Coefficient = 500)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-6 and Table 2.3-7 Table 2.3-3 and Table 2.3-7
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	20 minutes (Manual)
Control Room Occupancy Factor	Table 2.1-1

Table 3.5-2: Locked Rotor TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas Dose	5.2878E-01	1.5559E-01	4.4546E-01
Non-Noble Gas Dose - Iodine	9.1900E-01	5.1347E-01	2.9955E+00
Non-Noble Gas Dose – Alkali Metals	3.6098E-01	1.2094E-01	1.1508E+00
Initial SG Secondary Iodine	1.4821E-03	7.3853E-04	3.1709E-03
Control Room Shine			0.139
Total	**1.82**	**0.80**	**4.74**
Acceptance Limit	2.5	2.5	5

3.6 *Control Rod Ejection*

The Control Rod Ejection event involves a reactivity insertion that produces a short, rapid core power level increase which results in fuel rod damage and localized melting. For this event, a larger fraction of the core inventory is released from the damaged fuel than that identified in Table 2.2-6. Two separate release pathways are evaluated:, a release from containment and a release from the secondary system. In both cases, 10% of the

noble gases and 10% of the iodines in the core (Table 2.2-2) are available for release from the fuel gap of the damaged fuel rods. In addition, 12% of the alkali metals are also assumed to be located in the fuel rod gap.

For releases from containment, 10% of the fuel rods in the core are breached and 0.25% of the fuel experiences melting. The activity in the fuel rod gap of the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere. In addition, the gap inventory fractions are increased by a factor of 1.0104 to account for high burnup fuel as discussed in Sections 2.2.5 and 3.5. Moreover, 100% of the noble gases and 25% of the iodines in the melted fuel are also added to the fission product inventory in containment. No credit is taken for removal by containment sprays or for deposition of elemental iodine on containment surfaces. Natural deposition of aerosols in containment is assumed to occur beginning 24 hours after the start of the event. Activity is released from containment at the proposed Tech. Spec. leak rate. The release from the containment is based upon an atmospheric dispersion factor assuming a diffuse release from the containment surface. For conservatism, the release of iodine initially present in the steam generator secondary side is also considered to address any supplemental cooldown by the steam generators for this event.

For releases from the secondary system, 10% of the fuel rods in the core are breached and 0.25% of the fuel experiences melting. Activity released from the fuel is completely dissolved in the primary coolant and is available for release to the secondary system. The gap activity is increased by a factor of 1.0104 to address fuel rods with burnups that exceed the values of Footnote 11 of Reference [4.1]. In this case, 100% of the noble gases and 50% of the iodines in the melted fuel is also released into the reactor coolant. The noble gases are assumed to be released directly to the environment, and the remaining fission products are transported to the steam generators at the Tech. Spec. steam generator program leakage limit of 1 gpm. At the beginning of the event, a portion of the primary-to-secondary leakage is assumed to flash to vapor based upon the thermodynamic conditions of the reactor and secondary coolant, and the flashed leakage is released directly to the environment without mitigation. The unflashed portion of the tube leakage mixes with the bulk fluid in the steam generator secondary and becomes vapor at a rate that is a function of the steaming rate and the partition coefficients. After 40 minutes, the water level in the steam generator is assumed to fully cover the tube bundles, and all of the primary-to-secondary leakage is treated as unflashed. The leakage continues until steam releases are terminated when the RCS temperature is cooled to 212 °F at 24 hours. With the large amount of fission products introduced into the reactor coolant by failed fuel, the initial activity of the RCS prior to the event is not considered. However, the dose contribution from the iodine activity initially present in the steam generator secondary is included in the analysis. All releases from the secondary system occur from the PORVs/MSSVs.

The control room ventilation system is automatically realigned into the pressurization mode following receipt of a safety injection signal. Key control room ventilation parameters applied in the analysis are shown in Table 2.1-1. Other inputs and assumptions important to the Control Rod Ejection event are provided in Table 3.6-1. The dose consequences for this event are summarized in Table 3.6-2 and Table 3.6-3.

Table 3.6-1: Control Rod Ejection Inputs and Assumptions

Input/Assumption	Value
Source Term	Core Inventory (Table 2.2-2)
Fuel Rod Gap Fractions	Noble Gases - 0.10 Other Halogens - 0.10 Alkali Metals - 0.12
Percentage of Fuel Rods Failed	10%
Percentage of fuel that experience fuel melting	0.25%
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	150 rods in two assemblies
High Burnup multiplier applied to gap fractions	1.0104
Fuel Rod Peaking Factor	1.65
Initial Steam Generator Iodine Source Term	0.1 μCi/gm Dose Equivalent I-131
Iodine Chemical Form - Secondary Release	0% aerosol, 97% elemental, 3% organic
Iodine Chemical Form - Containment Release	95% aerosol, 4.85% elemental, 0.15% organic
Containment Volume	1,066,352 ft^3
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.18 %/day 0.09 %/day
Containment Leakage Filtration	0%
Natural Deposition in Containment	Elemental Iodine – None Aerosols - 0.1 hr^{-1} after 24 hours
Iodine/Particulate Removal by Containment Sprays	None
RCS Mass	466,141.5 lbm (minimum)
Steam Generator Secondary Liquid Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Primary-Secondary Leak Rate	1.0 gpm to all steam generators
Density Used for Leakage Volume-to-Mass Conversion	62.3 lbm/ft^3
Secondary Steam Release	0 - 2 hours: 460,000 lbm 2 - 8 hours: 1,256,000 lbm 8 - 24 hours: 1,347,000 lbm
Time to Cool RCS to 212 °F	24 hours
Duration of SG Tube Uncovery Following Reactor Trip	40 minutes

Input/Assumption	Value
Tube Leakage Flashing Fraction During Uncovery	0- 400 seconds: 8% 400-900 seconds: 6% 900-1700 seconds: 5.5% 1700 seconds-40 min: 4%
Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0
Steam Generator Moisture Carryover Fraction	0.2% (Particulate Partition Coefficient = 500)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-6 and Table 2.3-7 Table 2.3-3 and Table 2.3-7
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	120 seconds (Safety Injection)
Control Room Occupancy Factor	Table 2.1-1

Table 3.6-2: Control Rod Ejection Secondary Release TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas, Cladding Failure	9.6153E-01	2.8289E-01	8.0333E-01
Noble Gas, Fuel Melt	2.3793E-01	7.0005E-02	1.9879E-01
Non-Noble Gas, Cladding Failure, Iodine	1.1615E+00	6.5310E-01	1.8378E+00
Non-Noble Gas, Cladding Failure, Alkali	3.2816E-01	1.0995E-01	2.0428E-01
Non-Noble Gas, Fuel Melt	1.4520E-01	8.1647E-02	2.2976E-01
Initial SG Secondary Iodine	1.4821E-03	7.3853E-04	1.9539E-03
Control Room Shine			0.139
Total	**2.84**	**1.20**	**3.42**
Acceptance Limit	6.3	6.3	5

Table 3.6-3: Control Rod Ejection Containment Release TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Cladding Failure	3.8824E+00	2.6021E+00	1.5378E+00
Fuel Melt	1.8512E-01	1.1633E-01	6.8727E-02
Initial SG Secondary Iodine	1.4821E-03	7.3853E-04	1.9539E-03
Control Room Shine			0.139
Total	**4.07**	**2.72**	**1.75**
Acceptance Limit	6.3	6.3	5

3.7 *Waste Gas Decay Tank Rupture*

This event involves a major rupture of one of the Waste Gas Decay Tanks that causes the entire contents of the tank to be released directly to the environment. Reg. Guide 1.183 does not provide any guidance relative to the Waste Gas Decay Tank Rupture event. Guidelines for the WGDT analyses are given in Branch Technical Position 11-5 (BTP 11-5) of the Standard Review Plan (Reference [4.24]), with additional instruction available from Regulatory Issue Summary 2006-04 (Reference [4.2]). The activity in the tank is assumed to be equal to the noble gas content of the reactor coolant system during normal operation. This source term is derived from plant operation with 1% fuel defects which has operated for a sufficient length of time to achieve equilibrium radioactive concentrations in the RCS. The equilibrium RCS concentrations are then adjusted to 100/Ē as discussed in Section 2.2.3. For the WGDT rupture event, the entire noble gas inventory of the RCS is then assumed to be stripped and placed into a single tank. This inventory is conservatively calculated by taking the

noble gas specific activities in the RCS from Table 2.2-4 and multiplying by the maximum RCS mass as shown in Table 3.7-1. The total activity released is determined to be equal to 59,256.4 Curies Dose Equivalent Xe-133, which exceeds the single WGDT licensing limit of 43,800 Curies. The WGDT failure simulates a major tank rupture in which the entire contents of the tank are instantaneously released directly to the environment. No credit is taken for hold-up, dilution, or decay in the Auxiliary Building.

Section B.1.C of Reference [4.24] requires that the release to the environment occur through a pathway not normally used for planned releases and will require a reasonable time to detect and take remedial action to terminate the release. This condition is satisfied in the analysis by assuming that normal Auxiliary Building ventilation is not in service. As such, discharges to the plant vent are not ensured, and gases escaping from the ruptured tank are permitted to be released through building openings with the highest atmospheric dispersion factors. For the offsite dose, the most limiting atmospheric dispersion factor is from the north Auxiliary Building normal ventilation intake, and for the control room dose, the limiting release point is the south normal ventilation intake. The control room ventilation system is assumed to remain in the normal alignment since a safety injection signal, which is required to automatically place the system in the pressurization mode, is not received for this event.

Neither BTP 11-5 nor RIS 2006-04 require dose evaluations at the LPZ or for the control room for a WGDT failure; however, both evaluations are completed in this effort for consistency and completeness. Item 11 of RIS 2006-04 allows the use of existing current licensing basis acceptance criterion of 500 mrem whole body at the EAB when this event is not submitted as part of the AST implementation. Consequently, the results of the WGDT failure are evaluated against the existing 500 mrem acceptance criterion for both the EAB and LPZ. Reg. Guide 1.183 does identify that the criterion for the control room dose is provided in 10 CFR 50.67, which establishes the dose limit for the control room as 5 rem TEDE. This value is applied here.

A summary of the inputs and assumptions to the WGDT rupture analysis are given in Table 3.7-2, and the dose consequences are shown in Table 3.7-3.

Table 3.7-1: WGDT Source Term

Nuclide	RCS Activity (μCi/g)	Total Activity (Ci)
Kr-85m	5.204E-01	1.433E+02
Kr-85	2.385E+01	6.570E+03
Kr-87	3.299E-01	9.087E+01
Kr-88	9.148E-01	2.520E+02
Xe-131m	1.600E+00	4.407E+02
Xe-133m	1.423E+00	3.920E+02
Xe-133	1.037E+02	2.857E+04
Xe-135m	2.138E-01	5.889E+01
Xe-135	3.361E+00	9.258E+02
Xe-138	2.292E-01	6.314E+01

Table 3.7-2: WGDT Rupture Inputs and Assumptions

Input/Assumption	Value
Source Term	Table 3.7-1. Equal to 59,256.4 D.E. Xe-133
RCS Mass	607,290.6 lbm (275,460,950 gm) (maximum)
Tank Volume	500 ft^3 (arbitrary)
Tank Release Rate	1,000,000 cfm (conservatively high to simulate an instantaneous release)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-6 and Table 2.3-7 Table 2.3-3 and Table 2.3-7
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	Not Isolated
Control Room Occupancy Factor	Table 2.1-1

Table 3.7-3: WGDT Rupture Dose Results

Release	EAB (rem whole body)	LPZ (rem whole body)	Control Room (rem TEDE)
WGDT Rupture	0.22	0.04	0.09
Acceptance Limit	0.5	0.5	5

3.8 *Volume Control Tank Rupture*

The analysis of the VCT rupture conservatively assumes a failure of the VCT just prior to venting which releases the accumulated noble gases in the liquid and vapor phases of the tank. In addition, the noble gases within the fluid of the letdown line entering the tank continue to be released for an additional 15 minutes following the tank rupture. The tank and letdown line activities are calculated from RCS equilibrium noble gas concentrations based upon 1% failed fuel with no adjustment for 100/Ē. This conservative source term is consistent with Section B.1.B of Branch Technical Position 11-5. Accumulated gases in the VCT vapor space are calculated using gas stripping fractions, and the entire amount of dissolved gases in the VCT liquid space and the continued letdown flow are available for release after the tank ruptures. The resulting source term is shown in Table 3.8-1.

Further guidance for the analysis of this event is taken from BTP 11-5 since Reg. Guide 1.183 does not address waste system failures. Pathways for the noble gases to escape from the VCT to the environment include those which are not normally used for planned releases. For this event, the release location is the south normal ventilation intake for the control room dose, and is the north normal ventilation intake for the offsite doses based upon the most limiting atmospheric dispersion factors. Releases from the normal supply vents are possible when the Auxiliary Building ventilation system is not in service. This analytical approach is consistent with the guidance of Section B.1.C of BTP 11-5. The analysis assumes that this release is made directly to the environment, without taking credit for hold-up, dilution, or decay in the Auxiliary Building. Similarly, while the control room ventilation system filters would have no impact on this event, the control room ventilation system remains in the normal system alignment described in Section 2.1. Significant inputs and assumptions applied in the analysis of this event are listed in Table 3.8-2.

The acceptance criteria for this event is based upon the guidance of Item 11 of RIS 2006-04, which permits the use of the existing current licensing basis acceptance criterion of 500 mrem whole body at the EAB when AST is not implemented for this event. Without specific guidance for the LPZ, the 500 mrem whole body limit is also applied to the LPZ dose for this event. The acceptance limit of 5 rem TEDE to operators in the control room from 10 CFR 50.67 is applied based upon Section 4.4 of Reg. Guide 1.183. The dose consequences for the VCT rupture event are provided in Table 3.8-3.

Table 3.8-1: VCT Source Term

Nuclide	Equilibrium RCS (μCi/g)	VCT Gas Phase Activity (Ci)	VCT Liquid Phase Activity (Ci)	Letdown Flow Activity (Ci)	VCT Total Release Activity (Ci)
Kr-85m	1.366E+00	1.596E+02	1.021E+01	1.012E+01	1.799E+02
Kr-85	6.261E+01	1.835E+04	4.679E+02	4.639E+02	1.928E+04
Kr-87	8.660E-01	3.955E+01	6.472E+00	6.416E+00	5.244E+01
Kr-88	2.401E+00	2.070E+02	1.794E+01	1.779E+01	2.427E+02
Xe-131m	4.199E+00	8.716E+02	3.138E+01	3.111E+01	9.341E+02
Xe-133m	3.734E+00	7.125E+02	2.791E+01	2.766E+01	7.681E+02
Xe-133	2.723E+02	5.421E+04	2.035E+03	2.017E+03	5.826E+04
Xe-135m	5.612E-01	5.806E+00	4.194E+00	4.158E+00	1.416E+01
Xe-135	8.821E+00	1.200E+03	6.593E+01	6.535E+01	1.331E+03
Xe-138	6.017E-01	5.770E+00	4.497E+00	4.458E+00	1.473E+01

Table 3.8-2: VCT Rupture Inputs and Assumptions

Input/Assumption	Value
Source Term	Table 3.8-1
Tank Volume Liquid Volume	267 ft^3
Tank Volume Vapor Volume	500 ft^3 (arbitrary)
VCT Release Rate	1,000,000 cfm (conservatively high to simulate an instantaneous release)
Letdown Flow Rate	132 gpm
Letdown Isolation Time	15 minutes
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-6 and Table 2.3-7 Table 2.3-3 and Table 2.3-7
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-8.

Input/Assumption	Value
Control Room	3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	Not Isolated
Control Room Occupancy Factor	Table 2.1-1

Table 3.8-3: VCT Rupture Dose Results

Release	EAB (rem whole body)	LPZ (rem whole body)	Control Room (rem TEDE)
VCT Rupture	0.33	0.06	0.13
Acceptance Limit	0.5	0.5	5

3.9 *Results Summary*

The results of the D. C. Cook radiological analyses using the AST methodology are summarized in Table 3.9-1.

Table 3.9-1: Dose Results Summary

Event	EAB Dose (rem TEDE)	LPZ Dose (rem TEDE)	Control Room (rem TEDE)
LOCA	21.48	8.30	4.56
MSLB Pre-accident Iodine Spike	0.25	0.10	1.38
SGTR Pre-accident Iodine Spike	4.16	0.76	3.96
Acceptance Criteria	**25**	**25**	**5**
FHA – Containment Release	3.57	0.67	4.49
FHA – Auxiliary Building Release	0.68	0.14	0.27
Control Rod Ejection – Containment	4.07	2.72	1.75
Control Rod Ejection – Secondary	2.84	1.20	3.42
Acceptance Criteria	**6.3**	**6.3**	**5**
MSLB Concurrent Iodine Spike	0.84	0.29	2.99
SGTR Concurrent Iodine Spike	2.50(2)	0.47	2.26
Locked Rotor	1.82	0.80	4.74
Acceptance Criteria	**2.5**	**2.5**	**5**
WGDT Rupture	0.22(1)	0.04(1)	0.09
VCT Rupture	0.33(1)	0.06(1)	0.13
Acceptance Criteria	**0.5**(1)	**0.5**(1)	**5**

(1) EAB and LPZ results and acceptance criteria for the WGDT and VCT events are whole body doses
(2) Calculated value is 2.497 rem which is rounded up to 2.50.

4 References

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