APPENDIX 15A

EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

This appendix contains the parameters and models that form the basis of the radiological consequences analyses for the various postulated accidents.

15A.1 Offsite Dose Calculation Models

Radiological consequences analyses are performed to determine the total effective dose equivalent (TEDE) doses associated with the postulated accident. The determination of TEDE doses takes into account the committed effective dose equivalent (CEDE) dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the effective dose equivalent (EDE) dose resulting from immersion in the cloud of activity.

15A.1.1 Immersion Dose (Effective Dose Equivalent)

Assuming a semi-infinite cloud, the immersion doses are calculated using the equation:

$$D_{im} = \sum_i DCF_i \sum_j R_{ij} (\chi / Q)_j$$

where:

D_{im} = Immersion (EDE) dose (rem)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-s)

R_{ij} = Amount of isotope i released during time period j (Ci)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

15A.1.2 Inhalation Dose (Committed Effective Dose Equivalent)

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j R_{ij} (BR)_j (\chi / Q)_j$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

R_{ij} = Amount of isotope i released during time period j (Ci)

$(BR)_j$ = Breathing rate during time period j (m^3/s)

$(\chi/Q)_j$ = Atmospheric dispersion factor during time period j (s/m^3)

15A.1.3 Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.2 Main Control Room Dose Models

Radiological consequences analyses are performed to determine the TEDE doses associated with the postulated accident. The determination of TEDE doses takes into account the CEDE dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the EDE dose resulting from immersion in the cloud of activity.

15A.2.1 Immersion Dose Models

Due to the finite volume of air contained in the main control room, the immersion dose for an operator occupying the main control room is substantially less than it is for the case in which a semi-infinite cloud is assumed. The finite cloud doses are calculated using the geometry correction factor from Murphy and Campe (Reference 1).

The equation is:

$$D_{im} = \frac{1}{GF} \sum_i DCF_i \sum_j (IAR)_{ij} O_j$$

where:

D_{im} = Immersion (EDE) dose (rem)

GF = Main control room geometry factor
= $1173/V^{0.338}$

V = Volume of the main control room (ft^3)

DCF_i = EDE dose conversion factor for isotope i (rem-m^3/Ci-s)

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j (Ci-s/m^3)

O_j = Fraction of time period j that the operator is assumed to be present

15A.2.2 Inhalation Dose

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j (IAR)_{ij} (BR)_j O_j$$

where:

D_{CEDE} = CEDE dose (rem)

DCF_i = CEDE dose conversion factor (rem per curie inhaled) for isotope i

$(IAR)_{ij}$ = Integrated activity for isotope i in the main control room during time period j (Ci-s/m^3)

$(BR)_j$ = Breathing rate during time period j (m^3/s)

O_j = Fraction of time period j that the operator is assumed to be present

15A.2.3 Total Dose (Total Effective Dose Equivalent)

The TEDE doses are the sum of the EDE and the CEDE doses.

15A.3 General Analysis Parameters

15A.3.1 Source Terms

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the core if the accident involves fuel failures. The radiological consequences analyses use conservative design basis source terms.

15A.3.1.1 Primary Coolant Source Term

The design basis primary coolant source terms are listed in Table 11.1-2. These source terms are based on continuous plant operation with 0.25-percent fuel defects. The remaining assumptions used in determining the primary coolant source terms are listed in Table 11.1-1.

The accident dose analyses take into account increases in the primary coolant source terms for iodines and noble gases above those listed in Table 11.1-2, consistent with the Tech Spec limits of 1.0 μCi/g dose equivalent I-131 for the iodines and 280 μCi/g dose equivalent Xe-133 for the noble gases.

The radiological consequences analyses for certain accidents also take into account the phenomenon of iodine spiking, which causes the concentration of radioactive iodines in the primary coolant to increase significantly. Table 15A-1 lists the concentrations of iodine isotopes associated with a pre-existing iodine spike. This is an iodine spike that occurs prior to the accident and for which the peak primary coolant activity is reached at the time the accident is assumed to occur. These isotopic concentrations are also defined as 60 μCi/g dose equivalent I-131. The probability of this adverse timing of the iodine spike and accident is small.

Although it is unlikely for an accident to occur at the same time that an iodine spike is at its maximum reactor coolant concentration, for many accidents it is expected that an iodine spike would be initiated by the accident or by the reactor trip associated with the accident. Table 15A-2 lists the iodine appearance rates (rates at which the various iodine isotopes are transferred from the core to the primary coolant by way of the assumed cladding defects) for normal operation. The iodine spike appearance rates are assumed to be as much as 500 times the normal appearance rates.

15A.3.1.2 Secondary Coolant Source Term

The secondary coolant source term used in the radiological consequences analyses is conservatively assumed to be 10 percent of the primary coolant equilibrium source term. This is more conservative than using the design basis secondary coolant source terms listed in Table 11.1-5.

Because the iodine spiking phenomenon is short-lived and there is a high level of conservatism for the assumed secondary coolant iodine concentrations, the effect of iodine spiking on the secondary coolant iodine source terms is not modeled.

There is assumed to be no secondary coolant noble gas source term because the noble gases entering the secondary side due to primary-to-secondary leakage enter the steam phase and are discharged via the condenser air removal system.

15A.3.1.3 Core Source Term

Table 15A-3 lists the core source terms at shutdown for an assumed three-region equilibrium cycle at end of life after continuous operation at 2 percent above full core thermal power. The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative. In addition to iodines and noble gases, the source terms listed include nuclides that are identified as potentially significant dose contributors in the event of a degraded core accident. The design basis loss-of-coolant accident analysis is not expected to result in significant core damage, but the radiological consequences analysis assumes severe core degradation.

15A.3.2 Nuclide Parameters

The radiological consequence analyses consider radioactive decay of the subject nuclides prior to their release, but no additional decay is assumed after the activity is released to the environment. Table 15A-4 lists the decay constants for the nuclides of concern.

Table 15A-4 also lists the dose conversion factors for calculation of the CEDE doses due to inhalation of iodines and other nuclides and EDE dose conversion factors for calculation of the dose due to immersion in a cloud of activity. The CEDE dose conversion factors are from EPA

Federal Guidance Report No. 11 (Reference 2) and the EDE dose conversion factors are from EPA Federal Guidance Report No. 12 (Reference 3).

15A.3.3 Atmospheric Dispersion Factors

Subsection 2.3.4 lists the off-site short-term atmospheric dispersion factors (χ/Q) for the reference site. Table 15A-5 (Sheet 1 of 2) reiterates these χ/Q values.

The atmospheric dispersion factors (χ/Q) to be applied to air entering the main control room following a design basis accident are specified at the HVAC intake and at the annex building entrance (which would be the air pathway to the main control room due to ingress/egress). A set of χ/Q values is identified for each potential activity release location that has been identified and the two control room receptor locations. These χ/Q values are listed in Table 15A-6 and are provided in Table 2-1 (Sheet 3 of 3).

The site-specific control room χ/Q values shall be bounded by the values in Table 15A-6. For a site selected that has χ/Q values that exceed the values in Table 15A-6, how the radiological consequences associated with the controlling design basis accident continue to meet the control room operator dose limits given in General Design Criteria 19 using site-specific χ/Q values should be addressed. Topographical characteristics in the vicinity of the site for restrictions of horizontal and/or vertical plume spread, channeling or other changes in airflow trajectories, and other unusual conditions affecting atmospheric transport and diffusion between the source and the receptors should be considered. No further action is required for sites within the bounds of the site parameters for atmospheric dispersion.

Table 15A-7 identifies the AP1000 source and receptor data to be used when determining the site-specific control room χ/Q values using the ARCON96 code (References 4 and 5).

The main control room χ/Q values do not incorporate occupancy factors.

The locations of the potential release points and their relationship to the main control room air intake and the annex building access door are shown in Figure 15A-1.

15A.4 References

1. Murphy, K. G., Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," paper presented at the 13th AEC Air Cleaning Conference.

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

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

4. NUREG/CR-6331, Ramsdell, J. V. and Simonen, C. A., "Atmospheric Relative Concentrations in Building Wakes," Revision 1, May 1997.

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

Table 15A-1

REACTOR COOLANT IODINE CONCENTRATIONS FOR MAXIMUM IODINE SPIKE OF 60 µCi/g DOSE EQUIVALENT I-131

Nuclide	µCi/g
I-130	0.66
I-131	43.4
I-132	57.5
I-133	78.8
I-134	13.4
I-135	47.8

Table 15A-2

IODINE APPEARANCE RATES IN THE REACTOR COOLANT

Nuclide	Equilibrium Appearance Rate (Ci/min)
I-130	7.03×10^{-3}
I-131	3.39×10^{-1}
I-132	1.38
I-133	7.42×10^{-1}
I-134	6.81×10^{-1}
I-135	6.37×10^{-1}

Table 15A-3 (Sheet 1 of 2)

REACTOR CORE SOURCE TERM[(1)]

	Nuclide	Inventory (Ci)		Nuclide	Inventory (Ci)
Iodines	I-130	3.66×10^{6}	**Noble Gases**	Kr-85m	2.63×10^{7}
	I-131	9.63×10^{7}		Kr-85	1.06×10^{6}
	I-132	1.40×10^{8}		Kr-87	5.07×10^{7}
	I-133	1.99×10^{8}		Kr-88	7.14×10^{7}
	I-134	2.18×10^{8}		Xe-131m	1.06×10^{6}
	I-135	1.86×10^{8}		Xe-133m	5.84×10^{6}
Cs Group	Cs-134	1.94×10^{7}		Xe-133	1.90×10^{8}
	Cs-136	5.53×10^{6}		Xe-135m	3.87×10^{7}
	Cs-137	1.13×10^{7}		Xe-135	4.84×10^{7}
	Cs-138	1.82×10^{8}		Xe-138	1.65×10^{8}
	Rb-86	2.29×10^{5}	**Sr & Ba**	Sr-89	9.66×10^{7}
Te Group	Te-127m	1.32×10^{6}		Sr-90	8.31×10^{6}
	Te-127	1.02×10^{7}		Sr-91	1.20×10^{8}
	Te-129m	4.50×10^{6}		Sr-92	1.29×10^{8}
	Te-129	3.04×10^{7}		Ba-139	1.78×10^{8}
	Te-131m	1.40×10^{7}		Ba-140	1.71×10^{8}
	Te-132	1.38×10^{8}	**Ce Group**	Ce-141	1.63×10^{8}
	Sb-127	1.03×10^{7}		Ce-143	1.52×10^{8}
	Sb-129	3.10×10^{7}		Ce-144	1.23×10^{8}
Ru Group	Ru-103	1.45×10^{8}		Pu-238	3.83×10^{5}
	Ru-105	9.83×10^{7}		Pu-239	3.37×10^{4}
	Ru-106	4.77×10^{7}		Pu-240	4.94×10^{4}
	Rh-105	9.00×10^{7}		Pu-241	1.11×10^{7}
	Mo-99	1.84×10^{8}		Np-239	1.93×10^{9}
	Tc-99m	1.61×10^{8}			

Note:

1. The following assumptions apply:
   - Core thermal power of 3468 MWt (2 percent above the design core power of 3400 MWt). The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.
   - Three-region equilibrium cycle core at end of life

Table 15A-3 (Sheet 2 of 2)

REACTOR CORE SOURCE TERM[(1)]

	Nuclide	Inventory (Ci)
La Group	Y-90	8.66×10^6
	Y-91	1.24×10^8
	Y-92	1.30×10^8
	Y-93	1.49×10^8
	Nb-95	1.67×10^8
	Zr-95	1.66×10^8
	Zr-97	1.64×10^8
	La-140	1.82×10^8
	La-141	1.62×10^8
	La-142	1.57×10^8
	Pr-143	1.46×10^8
	Nd-147	6.48×10^7
	Am-241	1.25×10^4
	Cm-242	2.95×10^6
	Cm-244	3.62×10^5

Note:

1. The following assumptions apply:
   - Core thermal power of 3468 MWt (2 percent above the design core power of 3400 MWt). The main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.
   - Three-region equilibrium cycle core at end of life

Table 15A-4 (Sheet 1 of 4)

NUCLIDE PARAMETERS

A. HALOGENS

Isotope	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
I-130	$5.61x10^{-2}$	$1.04x10^{-13}$	$7.14x10^{-10}$
I-131	$3.59x10^{-3}$	$1.82x10^{-14}$	$8.89x10^{-9}$
I-132	$3.01x10^{-1}$	$1.12x10^{-13}$	$1.03x10^{-10}$
I-133	$3.33x10^{-2}$	$2.94x10^{-14}$	$1.58x10^{-9}$
I-134	$7.91x10^{-1}$	$1.30x10^{-13}$	$3.55x10^{-11}$
I-135	$1.05x10^{-1}$	$7.98x10^{-14}$	$3.32x10^{-10}$

B. NOBLE GASES

Isotope	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	
Kr-85m	$1.55x10^{-1}$	$7.48x10^{-15}$	
Kr-85	$7.38x10^{-6}$	$1.19x10^{-16}$	
Kr-87	$5.45x10^{-1}$	$4.12x10^{-14}$	
Kr-88	$2.44x10^{-1}$	$1.02x10^{-13}$	
Xe-131m	$2.43x10^{-3}$	$3.89x10^{-16}$	
Xe-133m	$1.32x10^{-2}$	$1.37x10^{-15}$	
Xe-133	$5.51x10^{-3}$	$1.56x10^{-15}$	
Xe-135m	2.72	$2.04x10^{-14}$	
Xe-135	$7.63x10^{-2}$	$1.19x10^{-14}$	
Xe-138	2.93	$5.77x10^{-14}$	

Table 15A-4 (Sheet 2 of 4)

NUCLIDE PARAMETERS

C. ALKALI METALS

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Cs-134	$3.84x10^{-5}$	$7.57x10^{-14}$	$1.25x10^{-8}$
Cs-136	$2.2x10^{-3}$	$1.06x10^{-13}$	$1.98x10^{-9}$
Cs-137(1)	$2.64x10^{-6}$	$2.88x10^{-14}$	$8.63x10^{-9}$
Cs-138	1.29	$1.21x10^{-13}$	$2.74x10^{-11}$
Rb-86	$1.55x10^{-3}$	$4.81x10^{-15}$	$1.79x10^{-9}$

D. TELLURIUM GROUP

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Te-127m	$2.65x10^{-4}$	$1.47x10^{-16}$	$5.81x10^{-9}$
Te-127	$7.41x10^{-2}$	$2.42x10^{-16}$	$8.60x10^{-11}$
Te-129m	$8.6x10^{-4}$	$1.55x10^{-15}$	$6.47x10^{-9}$
Te-129	$5.98x10^{-1}$	$2.75x10^{-15}$	$2.42x10^{-11}$
Te-131m	$2.31x10^{-2}$	$7.01x10^{-14}$	$1.73x10^{-9}$
Te-132	$8.86x10^{-3}$	$1.03x10^{-14}$	$2.55x10^{-9}$
Sb-127	$7.5x10^{-3}$	$3.33x10^{-14}$	$1.63x10^{-9}$
Sb-129	$1.6x10^{-1}$	$7.14x10^{-14}$	$1.74x10^{-10}$

E. STRONTIUM AND BARIUM

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Sr-89	$5.72x10^{-4}$	$7.73x10^{-17}$	$1.12x10^{-8}$
Sr-90	$2.72x10^{-6}$	$7.53x10^{-18}$	$3.51x10^{-7}$
Sr-91	$7.3x10^{-2}$	$3.45x10^{-14}$	$4.49x10^{-10}$
Sr-92	$2.56x10^{-1}$	$6.79x10^{-14}$	$2.18x10^{-10}$
Ba-139	$5.02x10^{-1}$	$2.17x10^{-15}$	$4.64x10^{-11}$
Ba-140	$2.27x10^{-3}$	$8.58x10^{-15}$	$1.01x10^{-9}$

Note:
1. The listed average gamma disintegration energy for Cs-137 is due to the production and decay of Ba-137m.

Table 15A-4 (Sheet 3 of 4)

NUCLIDE PARAMETERS

F. NOBLE METALS

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Ru-103	$7.35x10^{-4}$	$2.25x10^{-14}$	$2.42x10^{-9}$
Ru-105	$1.56x10^{-1}$	$3.81x10^{-14}$	$1.23x10^{-10}$
Ru-106	$7.84x10^{-5}$	0.0	$1.29x10^{-7}$
Rh-105	$1.96x10^{-2}$	$3.72x10^{-15}$	$2.58x10^{-10}$
Mo-99	$1.05x10^{-2}$	$7.28x10^{-15}$	$1.07x10^{-9}$
Tc-99m	$1.15x10^{-1}$	$5.89x10^{-15}$	$8.80x10^{-12}$

G. CERIUM GROUP

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Ce-141	$8.89x10^{-4}$	$3.43x10^{-15}$	$2.42x10^{-9}$
Ce-143	$2.1x10^{-2}$	$1.29x10^{-14}$	$9.16x10^{-10}$
Ce-144	$1.02x10^{-4}$	$8.53x10^{-16}$	$1.01x10^{-7}$
Pu-238	$9.02x10^{-7}$	$4.88x10^{-18}$	$1.06x10^{-4}$
Pu-239	$3.29x10^{-9}$	$4.24x10^{-18}$	$1.16x10^{-4}$
Pu-240	$1.21x10^{-8}$	$4.75x10^{-18}$	$1.16x10^{-4}$
Pu-241	$5.5x10^{-6}$	$7.25x10^{-20}$	$2.23x10^{-6}$
Np-239	$1.23x10^{-2}$	$7.69x10^{-15}$	$6.78x10^{-10}$

Table 15A-4 (Sheet 4 of 4)

NUCLIDE PARAMETERS

H. LANTHANIDE GROUP

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Y-90	$1.08x10^{-2}$	$1.90x10^{-16}$	$2.28x10^{-9}$
Y-91	$4.94x10^{-4}$	$2.60x10^{-16}$	$1.32x10^{-8}$
Y-92	$1.96x10^{-1}$	$1.30x10^{-14}$	$2.11x10^{-10}$
Y-93	$6.86x10^{-2}$	$4.80x10^{-15}$	$5.82x10^{-10}$
Nb-95	$8.22x10^{-4}$	$3.74x10^{-14}$	$1.57x10^{-9}$
Zr-95	$4.51x10^{-4}$	$3.60x10^{-14}$	$6.39x10^{-9}$
Zr-97	$4.1x10^{-2}$	$9.02x10^{-15}$	$1.17x10^{-9}$
La-140	$1.72x10^{-2}$	$1.17x10^{-13}$	$1.31x10^{-9}$
La-141	$1.76x10^{-1}$	$2.39x10^{-15}$	$1.57x10^{-10}$
La-142	$4.5x10^{-1}$	$1.44x10^{-13}$	$6.84x10^{-11}$
Nd-147	$2.63x10^{-3}$	$6.19x10^{-15}$	$1.85x10^{-9}$
Pr-143	$2.13x10^{-3}$	$2.10x10^{-17}$	$2.19x10^{-9}$
Am-241	$1.83x10^{-7}$	$8.18x10^{-16}$	$1.20x10^{-4}$
Cm-242	$1.77x10^{-4}$	$5.69x10^{-18}$	$4.67x10^{-6}$
Cm-244	$4.37x10^{-6}$	$4.91x10^{-18}$	$6.70x10^{-5}$

Table 15A-5

OFFSITE ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

Dose Location	LOCA Dose Analysis	Other Dose Analyses(1)
Site boundary χ/Q (s/m^3)		
0 – 2 hours(2)	5.1×10^{-4}	1.0×10^{-3}
Low population zone χ/Q (s/m^3)		
0 – 8 hours	2.2×10^{-4}	5.0×10^{-4}
8 – 24 hours	1.6×10^{-4}	3.0×10^{-4}
24 – 96 hours	1.0×10^{-4}	1.5×10^{-4}
96 – 720 hours	8.0×10^{-5}	8.0×10^{-5}

Notes:
1. For all design basis accidents analyzed in Chapter 15 other than LOCA
2. Nominally defined as the 0- to 2-hour interval, but is applied to the 2-hour interval having the highest activity releases in order to address 10 CFR Part 50.34 requirements

Table 15A-6

CONTROL ROOM ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points[(1)]						
	Plant Vent or PCS Air Diffuser[(3)]	**Ground Level Containment Release Points[(4)]**	**PORV and Safety Valve Releases[(5)]**	**Steam Line Break Releases**	**Fuel Handling Area[(6)]**	**Condenser Air Removal Stack[(7)]**
0 – 2 hours	3.0E-3	6.0E-3	2.0E-2	2.4E-2	6.0E-3	6.0E-3
2 – 8 hours	2.5E-3	3.6E-3[(8)]	1.8E-2	2.0E-2	4.0E-3	4.0E-3
8 – 24 hours	1.0E-3	1.4E-3[(8)]	7.0E-3	7.5E-3	2.0E-3	2.0E-3
1 – 4 days	8.0E-4	1.8E-3	5.0E-3	5.5E-3	1.5E-3	1.5E-3
4 – 30 days	6.0E-4	1.5E-3	4.5E-3	5.0E-3	1.0E-3	1.0E-3
χ/Q (s/m^3) at Annex Building Door for the Identified Release Points[(2)]						
	Plant Vent or PCS Air Diffuser[(3)]	**Ground Level Containment Release Points[(4)]**	**PORV and Safety Valve Releases[(5)]**	**Steam Line Break Releases**	**Fuel Handling Area[(6)]**	**Condenser Air Removal Stack[(7)]**
0 – 2 hours	1.0E-3	1.0E-3	4.0E-3	4.0E-3	6.0E-3	2.0E-2
2 – 8 hours	7.5E-4	7.5E-4	3.2E-3	3.2E-3	4.0E-3	1.8E-2
8 – 24 hours	3.5E-4	3.5E-4	1.2E-3	1.2E-3	2.0E-3	7.0E-3
1 – 4 days	2.8E-4	2.8E-4	1.0E-3	1.0E-3	1.5E-3	5.0E-3
4 – 30 days	2.5E-4	2.5E-4	8.0E-4	8.0E-4	1.0E-3	4.5E-3

Notes:

1. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the non-safety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.
2. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.
3. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss-of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.
4. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.

5. The listed values bound the dispersion factors for releases from the steam line safety & power-operated relief valves. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and for the secondary side release from a rod ejection accident.
6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel building relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.
7. This release point is included for information only as a potential activity release point. None of the design basis accident radiological consequences analyses model release from this point.
8. The LOCA dose analysis models the ground level containment release point HVAC intake atmospheric dispersion factors. The other dose analyses consider atmospheric dispersion factors of 4.5E-3 s/m^3 for the 2- to 8-hour interval and 2.0E-3 s/m^3 for the 8- to 24-hour interval.

Table 15A-7

CONTROL ROOM SOURCE/RECEPTOR DATA FOR DETERMINATION OF ATMOSPHERIC DISPERSION FACTORS

Source Description	Release Elevation Note 1 (m)	Horizontal Straight-Line Distance To Receptor		
		Control Room HVAC Intake (Elevation 19.9 m) (Δ1)	Annex Building Access (Elevation 1.5 m) (Δ2)	Comment
Plant Vent (⬡1)	55.7	147.2 ft (44.9 m)	379.3 ft (115.6 m)	
PCS Air Diffuser (⬡2)	69.8	118.1 ft (36.0 m)	343.2 ft (104.6 m)	
Fuel Building Blowout Panel (⬡3)	17.4	203.2 ft (61.9 m)	427.4 ft (130.3 m)	Note 3
Radwaste Building Truck Staging Area Door (⬡4)	1.5	218.5 ft (66.6 m)	433.5 ft (132.1 m)	Note 3
Steam Vent (⬡5)	17.1	61.5 ft (18.8 m)	261.6 ft (79.7 m)	
PORV/Safety Valves (⬡6)	19.2	66.9 ft (20.4 m)	255.4 ft (77.8 m)	
Condenser Air Removal Stack (⬡7)	38.4	198.3 ft (60.4 m)	58.3 ft (17.8 m)	Note 3
Containment Shell (Diffuse Area Source) (⬡8)	Same as Receptor Elevation (19.9 m or 1.5 m)	42.0 ft (12.8 m)	272.3 ft (83.0 m)	Note 2

Notes:
1. All elevations relative to grade at 0.0 m.
2. For calculating distance, the source is defined as the point on the containment shell closest to receptor.
3. Vertical distance traveled is conservatively neglected.
4. ⬡ – Refer to Symbols on Figure 15A-1.
5. Δ – Refer to Symbols on Figure 15A-1.

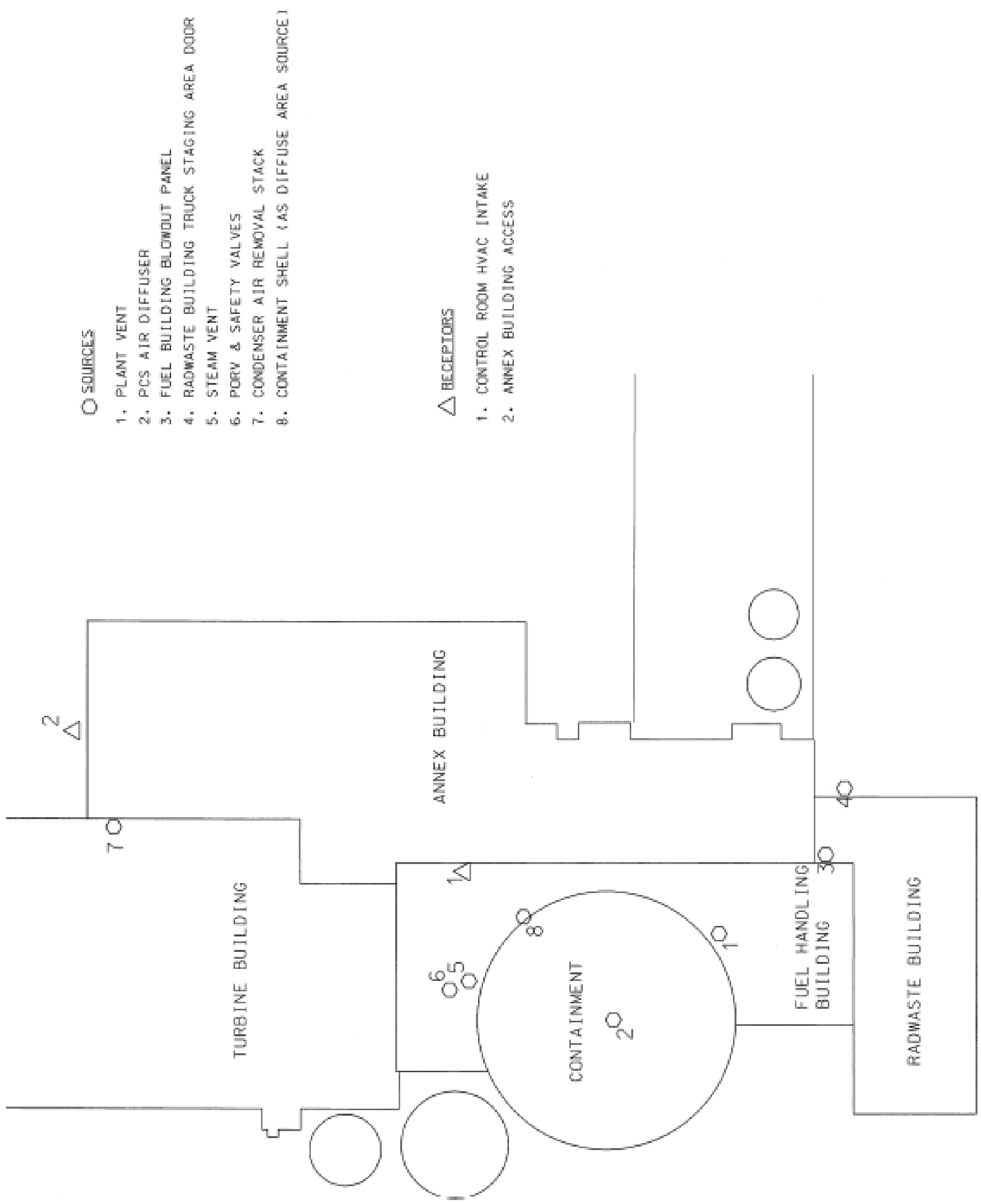

Figure 15A-1

Site Plan with Release and Intake Locations