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 acute dose resulting from immersion in the cloud of activity.

15A.1.1 Immersion Dose Models

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

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

where:

$D_{im(g)}$ = Noble gas immersion dose (rem)

DCF_i = Whole body dose conversion factor for noble gas isotope i (rem-m^3/Ci-s)

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

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

The acute dose resulting from immersion in the semi-infinite cloud of nongaseous activity is also calculated. The determination uses the equation:

$$D_{im(p)} = (0.25) \sum_i \overline{E}_{\gamma i} \sum_j R_{ij} \; (\chi/Q)_j$$

where:

$D_{im(p)}$ = Particulates immersion dose (rem)

$\overline{E}_{\gamma i}$ = Average gamma disintegration energy for isotope i (Mev/disintegration)

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 (rem)

DCF_i = CEDE 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 immersion 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 acute 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 doses 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). There are two models used: one for the noble gases and a second one for other airborne activity.

The equation for noble gases is:

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

where:

$D_{im(g)}$ = Noble gas immersion dose (rem)

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

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

DCF_i = Whole body 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

The acute dose resulting from immersion in the cloud of nongaseous activity is calculated using:

$$D_{im(p)} = \frac{0.25}{GF} \sum_i \overline{E}_{\gamma i} \sum_j (IAR)_{ij}\ O_j$$

where:

$D_{im(p)}$ = Particulates immersion dose (rem)

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

V = Volume of the main control room

$\overline{E}_{\gamma i}$ = Average gamma disintegration energy for isotope i (Mev/disintegration)

$(IAR)_{ij}$ = 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 immersion 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 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 24 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 both normal operation and for the iodine spike. The iodine spike appearance rates are assumed to be 500 times the normal appearance rates. The appearance rates are assumed to return to normal after the primary coolant activity reaches the maximum spike concentrations listed in Table 15A-1.

15A.3.1.2 Secondary Coolant Source Term

The secondary coolant iodine source term used in the radiological consequences analyses is conservatively assumed to be 10 percent of the design basis primary coolant source term provided in Table 11.1-2. 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 core source terms are based on "high burnup" fuel management with an average discharge region burnup of 55,080 MWD/MTU. 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.

For accidents involving assumed fuel cladding damage, the activity available for release is limited to the iodines, cesiums, and noble gases present in the fuel-cladding gap. Based on NUREG-1465 (Reference 2), the gap fraction is 3 percent of the core inventory if long-term fuel cooling is maintained. However, NUREG-1465 also refers to uncertainty regarding the applicability of this gap fraction to fuel having burnups greater than 40,000 MWD/MTU. The gap fraction is conservatively increased by 20 percent to address this uncertainty. The resulting gap fraction of 3.6 percent is used for the fuel handling accident, the rod ejection accident, and the locked reactor coolant pump rotor accident.

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, dose conversion factors for calculation of the acute dose due to immersion in a cloud of noble gases, and average gamma disintegration energies for determination of acute dose due to immersion in a cloud of nongaseous activity. These dose conversion factors are from EPA Federal Guidance Report No. 11 (Reference 3) and the average gamma disintegration energies are from ICRP Publication 38 (Reference 4).

15A.3.3 Atmospheric Dispersion Factors

Subsection 2.3.4 lists the short-term atmospheric dispersion factors (χ/Q) for the reference site. Table 15A-5 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 were calculated 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). As recommended by the NRC, the calculation of χ/Q values was performed using the computer code ARCON96 (Reference 5), which provides a time-based building wake model for the determination of χ/Q. The χ/Q values were calculated using three separate meteorological data bases to encompass a range of potential plant sites. Three existing power plant sites were used: a seacoast site, a river valley site, and a rolling-hills site. Each site data base included five years of meteorological data.

To address the uncertainty regarding the actual orientation of the AP600 at a site, the calculation of χ/Q was performed with the plant orientation ranging through the 16 compass points (every 22.5 degrees).

The ARCON96 code was run for all combinations of source and reception points for each of the 16 plant orientations at each of the three sites. These runs produced the 95th percentile χ/Q values for the five post-accident time periods of interest. Additional conservatism was added to the results for each site by selecting the maximum 95th percentile χ/Q value from the 16 compass points and then selecting the maximum of the three site values for application to the AP600 reference site. The AP600 reference site thus, for each time period, selectively combines the most conservative of the calculated values for the three sites and the 16 plant orientations.

Further adjustment was made to the main control room χ/Q values to add margin to the calculated values. This adjustment decreases the likelihood that a selected site would have higher χ/Q values than identified in this application, while still meeting the dose guidelines for the main control room. The χ/Q values are provided in Table 15A-5.

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

The locations of the potential release points for a Loss-of-Coolant Accident and their relationship to the main control room air intake and the personnel access door are shown in Figure 15A-1. Figure 15A-2 shows the locations of the potential release points associated with other postulated accidents relative to the possible paths for air entry into the main control room.

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. Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

3. 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.

4. ICRP Publication 38, "Radionuclide Transformations – Energy and Intensity of Emissions," 1983.

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

Table 15A-1

REACTOR COOLANT IODINE CONCENTRATIONS FOR MAXIMUM IODINE SPIKE OF 24 μCi/g DOSE EQUIVALENT I-131

Nuclide	μCi/g
I-131	17.2
I-132	29.5
I-133	35.6
I-134	6.1
I-135	19.7

Table 15A-2

IODINE APPEARANCE RATES IN THE REACTOR COOLANT(1)

Nuclide	Equilibrium Appearance Rate (Ci/min)	Iodine Spike Appearance Rate(Ci/min)
I-131	0.102	51
I-132	0.549	274
I-133	0.252	126
I-134	0.237	118
I-135	0.202	101

Note:

1. Appearance rates are based on a reactor coolant mass of 1.54×10^8 grams.

Table 15A-3 (Sheet 1 of 2)

REACTOR CORE SOURCE TERM[1]

	Nuclide	Inventory (Ci)		Nuclide	Inventory (Ci)
Iodines	I-131	5.2×10^7	**Noble Gases**	Kr-85m	1.4×10^7
	I-132	7.5×10^7		Kr-85	7.2×10^5
	I-133	1.1×10^8		Kr-87	2.7×10^7
	I-134	1.2×10^8		Kr-88	3.8×10^7
	I-135	1.0×10^8		Xe-131m	5.6×10^5
				Xe-133m	3.3×10^6
Cs Group	Cs-134	1.1×10^7		Xe-133	1.1×10^8
	Cs-136	3.6×10^6		Xe-135m	2.1×10^7
	Cs-137	7.8×10^6		Xe-135	3.3×10^7
	Rb-86	1.2×10^5		Xe-138	8.9×10^7
Te Group	Te-127m	7.3×10^5	**Sr & Ba**	Sr-89	5.2×10^7
	Te-127	5.4×10^6		Sr-90	5.8×10^6
	Te-129m	2.4×10^6		Sr-91	6.5×10^7
	Te-129	1.6×10^7		Sr-92	7.0×10^7
	Te-131m	7.4×10^6		Ba-139	9.6×10^7
	Te-132	7.4×10^7		Ba-140	9.2×10^7
	Sb-127	5.4×10^6			
	Sb-129	1.6×10^7	**Ce Group**	Ce-141	8.7×10^7
				Ce-143	8.1×10^7
Ru Group	Ru-103	7.9×10^7		Ce-144	7.0×10^7
	Ru-105	5.3×10^7		Pu-238	2.0×10^5
	Ru-106	2.9×10^7		Pu-239	2.2×10^4
	Rh-105	4.9×10^7		Pu-240	3.0×10^4
	Mo-99	9.6×10^7		Pu-241	7.9×10^6
	Tc-99m	8.4×10^7		Np-239	9.8×10^8

<u>Note:</u>

1. The following assumptions apply:

- Core thermal power of 1972 MWt (2 percent above the design core power of 1933 MWt)
- 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	6.0×10^{6}
	Y-91	6.7×10^{7}
	Y-92	7.0×10^{7}
	Y-93	8.0×10^{7}
	Nb-95	9.1×10^{7}
	Zr-95	9.0×10^{7}
	Zr-97	8.8×10^{7}
	La-140	9.5×10^{7}
	La-141	8.7×10^{7}
	La-142	8.5×10^{7}
	Pr-143	8.1×10^{7}
	Nd-147	3.5×10^{7}
	Am-241	1.0×10^{4}
	Cm-242	2.9×10^{6}
	Cm-244	2.2×10^{5}

Note:

1. The following assumptions apply:

- Core thermal power of 1972 MWt (2 percent above the design core power of 1933 MWt)
- Three-region equilibrium cycle core at end of life

Table 15A-4 (Sheet 1 of 4)

NUCLIDE PARAMETERS

A. IODINES

Isotope	Decay Constant (hr^{-1})	Average Gamma Disintegration Energy (Mev/disintegration)	CEDE Dose Conversion Factor (rem/Ci)
I-131	3.59×10^{-3}	3.8×10^{-1}	3.29×10^{4}
I-132	3.03×10^{-1}	2.24	3.81×10^{2}
I-133	3.33×10^{-2}	6.02×10^{-1}	5.85×10^{3}
I-134	7.91×10^{-1}	2.57	1.31×10^{2}
I-135	1.05×10^{-1}	1.55	1.23×10^{3}

B. NOBLE GASES

Isotope	Decay Constant (hr^{-1})	Dose Conversion Factor Immersion (rem-m^3/Ci-s)
Kr-85m	1.55×10^{-1}	3.07×10^{-2}
Kr-85	7.37×10^{-6}	4.84×10^{-4}
Kr-87	5.47×10^{-1}	1.46×10^{-1}
Kr-88	2.48×10^{-1}	3.70×10^{-1}
Xe-131m	2.41×10^{-3}	1.52×10^{-3}
Xe-133m	1.30×10^{-2}	5.53×10^{-3}
Xe-133	5.46×10^{-3}	6.24×10^{-3}
Xe-135m	2.72	7.75×10^{-2}
Xe-135	7.56×10^{-2}	4.82×10^{-2}
Xe-138	2.93	1.98×10^{-1}

Table 15A-4 (Sheet 2 of 4)

NUCLIDE PARAMETERS

C. CESIUM GROUP

Nuclide	Decay Constant (hr^{-1})	Average Gamma Disintegration Energy (Mev/disintegration)	CEDE Dose Conversion Factor (rem/Ci)
Cs-134	3.84×10^{-5}	1.55	4.62×10^{4}
Cs-136	2.2×10^{-3}	2.16	7.33×10^{3}
Cs-137(1)	2.64×10^{-6}	5.64×10^{-1}	3.19×10^{4}
Rb-86	1.55×10^{-3}	9.45×10^{-2}	6.63×10^{3}

D. TELLURIUM GROUP

Nuclide	Decay Constant (hr^{-1})	Average Gamma Disintegration Energy (Mev/disintegration)	CEDE Dose Conversion Factor (rem/Ci)
Te-127m	2.65×10^{-4}	1.12×10^{-2}	2.15×10^{4}
Te-127	7.41×10^{-2}	4.86×10^{-3}	3.18×10^{2}
Te-129m	8.6×10^{-4}	3.75×10^{-2}	2.39×10^{4}
Te-129	6.0×10^{-1}	5.91×10^{-2}	9.0×10^{1}
Te-131m	2.31×10^{-2}	1.42	6.4×10^{3}
Te-132	8.86×10^{-3}	2.33×10^{-1}	9.44×10^{3}
Sb-127	7.5×10^{-3}	6.88×10^{-1}	6.04×10^{3}
Sb-129	1.6×10^{-1}	1.44	6.44×10^{2}

E. STRONTIUM AND BARIUM

Nuclide	Decay Constant (hr^{-1})	Average Gamma Disintegration Energy (Mev/disintegration)	CEDE Dose Conversion Factor (rem/Ci)
Sr-89	5.72×10^{-4}	8.45×10^{-5}	4.14×10^{4}
Sr-90	2.72×10^{-6}	0.0	1.3×10^{6}
Sr-91	7.3×10^{-2}	6.93×10^{-1}	1.66×10^{3}
Sr-92	2.56×10^{-1}	1.34	8.1×10^{2}
Ba-139	5.02×10^{-1}	4.3×10^{-2}	1.7×10^{2}
Ba-140	2.27×10^{-3}	1.82×10^{-1}	3.74×10^{3}

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. RUTHENIUM GROUP

Nuclide	Decay Constant (hr^{-1})	Average Gamma Disintegration Energy (Mev/disintegration)	CEDE Dose Conversion Factor (rem/Ci)
Ru-103	7.35×10^{-4}	4.68×10^{-1}	8.95×10^{3}
Ru-105	1.56×10^{-1}	7.75×10^{-1}	4.55×10^{2}
Ru-106	7.84×10^{-5}	0.0	4.77×10^{5}
Rh-105	1.96×10^{-2}	7.8×10^{-2}	9.56×10^{2}
Mo-99	1.05×10^{-2}	1.5×10^{-1}	3.96×10^{3}
Tc-99m	1.15×10^{-1}	1.26×10^{-1}	3.3×10^{1}

G. CERIUM GROUP

Nuclide	Decay Constant (hr^{-1})	Average Gamma Disintegration Energy (Mev/disintegration)	CEDE Dose Conversion Factor (rem/Ci)
Ce-141	8.89×10^{-4}	7.6×10^{-2}	8.96×10^{3}
Ce-143	2.1×10^{-2}	2.82×10^{-1}	3.39×10^{3}
Ce-144	1.02×10^{-4}	2.1×10^{-2}	3.74×10^{5}
Pu-238	9.02×10^{-7}	1.81×10^{-3}	3.92×10^{8}
Pu-239	3.29×10^{-9}	8.08×10^{-4}	4.3×10^{8}
Pu-240	1.21×10^{-8}	1.73×10^{-3}	4.3×10^{8}
Pu-241	5.5×10^{-6}	2.54×10^{-6}	8.26×10^{6}
Np-239	1.23×10^{-2}	1.72×10^{-1}	2.51×10^{3}

Table 15A-4 (Sheet 4 of 4)

NUCLIDE PARAMETERS

H. LANTHANIDE GROUP

Nuclide	Decay Constant (hr^{-1})	Average Gamma Disintegration Energy (Mev/disintegration)	CEDE Dose Conversion Factor (rem/Ci)
Y-90	1.16×10^{-2}	1.7×10^{-6}	8.44×10^{3}
Y-91	4.94×10^{-4}	3.61×10^{-3}	4.89×10^{4}
Y-92	1.96×10^{-1}	2.51×10^{-1}	7.8×10^{2}
Y-93	6.86×10^{-2}	8.89×10^{-2}	2.15×10^{3}
Nb-95	8.22×10^{-4}	7.66×10^{-1}	5.81×10^{3}
Zr-95	4.51×10^{-4}	7.39×10^{-1}	2.37×10^{4}
Zr-97	4.1×10^{-2}	1.79×10^{-1}	4.33×10^{3}
La-140	1.72×10^{-2}	2.31	4.85×10^{3}
La-141	1.76×10^{-1}	4.27×10^{-2}	5.8×10^{2}
La-142	4.5×10^{-1}	2.68	2.5×10^{2}
Nd-147	2.63×10^{-3}	1.4×10^{-1}	6.85×10^{3}
Pr-143	2.13×10^{-3}	8.9×10^{-9}	1.09×10^{4}
Am-241	1.83×10^{-7}	3.24×10^{-2}	4.44×10^{8}
Cm-242	1.77×10^{-4}	1.83×10^{-3}	1.73×10^{7}
Cm-244	4.37×10^{-6}	1.7×10^{-3}	2.48×10^{8}

Table 15A-5 (Sheet 1 of 2)

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

Site boundary χ/Q (s/m^3)	
0 – 2 hours[1]	1.0×10^{-3}
Low population zone χ/Q (s/m^3)	
0 – 8 hours	1.35×10^{-4}
8 – 24 hours	1.0×10^{-4}
24 – 96 hours	5.4×10^{-5}
96 – 720 hours	2.2×10^{-5}

Note:

1. 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-5 (Sheet 2 of 2)

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

Main control room

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points(1)

	Elevated Containment Release(3)	Ground Level Containment Release Points(4)	Secondary Side Release Points(5)	Fuel Handling Area(6)	Fuel Building Relief Panel(7)
0 - 2 hours	1.2E-3	2.0E-3	2.0E-2	2.0E-3	3.0E-3
2 - 8 hours	8.0E-4	1.0E-3	1.8E-2	1.5E-3	2.0E-3
8 - 24 hours	4.0E-4	5.0E-4	8.0E-3	8.0E-4	1.0E-3
1 - 4 days	4.0E-4	5.0E-4	7.0E-3	8.0E-4	1.0E-3
4 - 30 days	3.0E-4	4.0E-4	6.0E-3	7.0E-4	9.0E-4

χ/Q (s/m^3) at Control Room Door for the Identified Release Points(2)

	Elevated Containment Release(3)	Ground Level Containment Release Points(4)	Secondary Side Release Points(5)	Fuel Handling Area(6)	Fuel Building Relief Panel(7)
0 - 2 hours	4.0E-4	1.0E-3	2.5E-3	1.0E-3	1.0E-3
2 - 8 hours	2.0E-4	6.0E-4	2.0E-3	6.0E-4	6.0E-4
8 - 24 hours	1.0E-4	3.0E-4	1.0E-4	3.0E-4	3.0E-4
1 - 4 days	9.0E-5	3.0E-4	9.0E-4	3.0E-4	3.0E-4
4 - 30 days	8.0E-5	3.0E-4	8.0E-4	2.5E-4	2.5E-4

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 apply to releases from the plant vent.

4. The listed values bound the dispersion factors for releases from the main equipment hatch and the staging area hatch. These dispersion factors would be 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 vents, the steam line safety & power-operated relief valves, and the condenser air removal stack. 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. Additionally, these dispersion coefficients are conservative for the small line break outside containment.

6. The listed values bound the dispersion factors for releases from the fuel storage and handling area. These dispersion factors would be used for the fuel handling accident occurring outside containment.

7. The listed values 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 to be used for evaluating the impact of releases associated with spent fuel pool boiling.

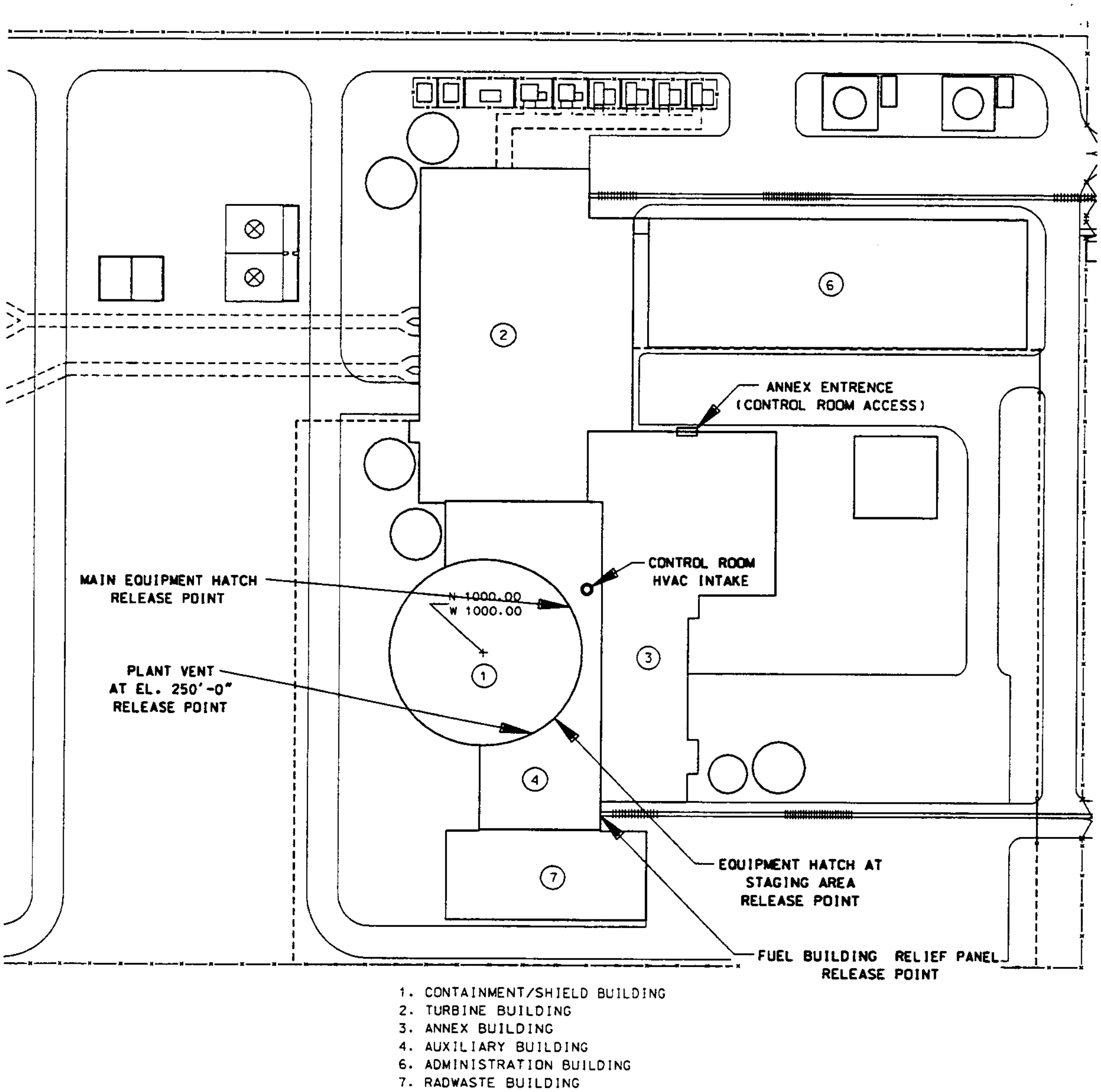
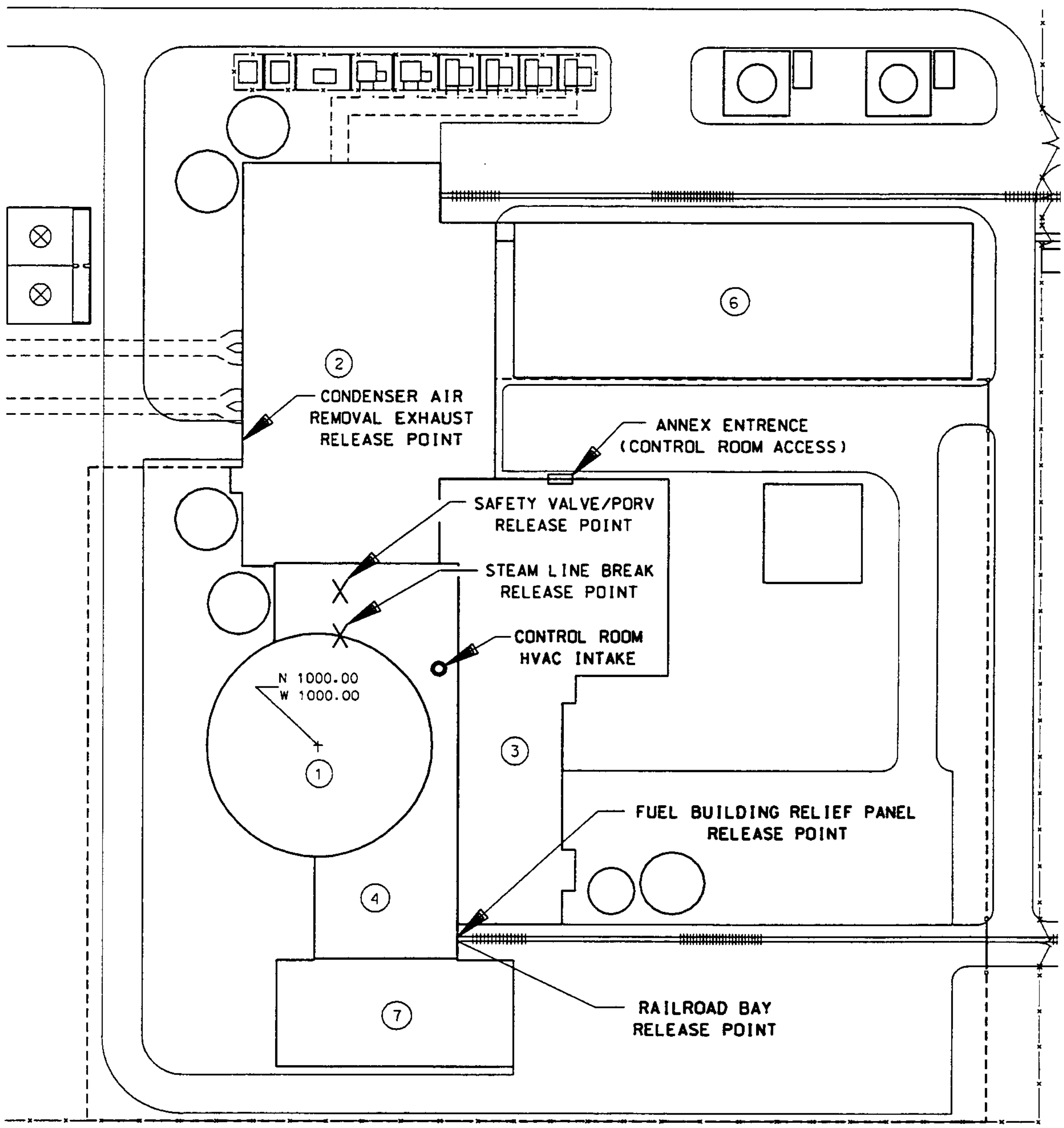


Figure 15A-1

Site Plan With Release and Intake Locations (LOCA Cases)



1. CONTAINMENT/SHIELD BUILDING
2. TURBINE BUILDING
3. ANNEX BUILDING
4. AUXILIARY BUILDING
6. ADMINISTRATION BUILDING
7. RADWASTE BUILDING

Figure 15A-2

Site Plan With Release and Intake Locations (Non-LOCA Cases)