Appendix D
Radioactive Source Bases

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Appendix D: Radioactive Source Bases

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Appendix D
Radioactive Source Bases

This appendix presents the quantities of radioactive isotopes present in the core, the fuel rod gap, the coolant, the volume control tank, and a gas decay tank. Activity calculations are consistent with the model and data from the ORIGEN2 computer code (Reference 2). Dose calculation models used in the safeguards analysis are presented. A general discussion of the derivations is also provided.

D.1 TOTAL ACTIVITY IN THE CORE

Numerical values for certain significant isotopes are given in Table D.3-2.

Table D.1-1
Core Total Fission Product Activities
Based on 1782.6 MWt (100.6% of 1772 MWt)

Isotope	Activity (Ci)
I-131	4.76E7
I-132	6.91E7
I-133	9.83E7
I-134	1.08E8
I-135	9.18E7
Kr-85m	1.31E7
Kr-85	5.39E5
Kr-87	2.53E7
Kr-88	3.56E7
Xe-131m	5.32E5
Xe-133m	2.88E6
Xe-133	9.42E7
Xe-135m	1.91E7
Xe-135	2.61E7
Xe-138	8.16E7
Cs-134	9.26E6
Cs-136	2.64E6
Cs-237	5.75E6
Rb-86	1.05E5
Te-127m	6.51E5
Te-127	5.01E6
Te-129m	2.22E6
Te-129	1.50E7
Te-131m	6.90E6
Te-132	6.80E7
Sb-127	5.05E6

Table D.1-1
Core Total Fission Product Activities
Based on 1782.6 MWt (100.6% of 1772 MWt)

Isotope	Activity (Ci)
Sb-129	1.53E7
Sr-89	4.82E7
Sr-90	4.26E6
Sr-91	5.97E7
Sr-92	6.44E7
Ba-139	8.81E7
Ba-140	8.48E7
Ru-103	7.16E7
Ru-105	4.81E7
Ru-106	2.38E7
Rh-105	4.45E7
Mo-99	9.08E7
Tc-99m	7.96E7
Ce-141	8.06E7
Ce-143	7.52E7
Ce-144	6.17E7
Pu-238	1.79E5
Pu-239	1.82E4
Pu-240	2.52E4
Pu-241	5.89E6
Np-239	9.50E8
Y-90	4.42E6
Y-91	6.18E7
Y-92	6.47E7
Y-93	7.42E7
Nb-95	8.27E7

Table D.1-1
Core Total Fission Product Activities
Based on 1782.6 MWt (100.6% of 1772 MWt)

Isotope	Activity (Ci)
Zr-95	8.22E7
Zr-97	8.13E7
La-140	9.21E7
La-141	8.05E7
La-142	7.80E7
Nd-147	3/21E7
Pr-143	7.27E7
Am-241	7.13E3
Cm-242	1.53E6
Cm-244	1.57E5

D.2 ACTIVITY IN THE FUEL GAP

The dose analyses model the activity that is in the gap using the fractions recommended in Regulatory Guide 1.183.

D.3 FUEL HANDLING SOURCES

The inventory of fission products in a fuel assembly is dependent on the power produced in the assembly. The fuel handling accident source term has been developed assuming a conservative power level, peaking factors and gap fractions. The parameters used in developing the conservative source term are summarized in Table D.3-1. The average fuel assembly activities following 100 hours of decay following shutdown are given in Table D.3-2.

Table D.3-1
Assumptions for Determining the Source Term for a Fuel Handling Accident

Parameter	Value
Reactor power (Mw_{th})	1782-6
Radial peaking value	1.70
Heavy metal loading value (kgU)	406
U-235 weight percent (wt%)	4.5
Cycle burnup (GWD/MTU)	17.8
Gap Fractions	
Iodine-131	0.08
Kr-85	0.10
All other iodine and noble gases	0.05
Number of assemblies damaged	1
Number of rodlets per assembly	179
Decay period since shutdown (hours)	100

Table D.3-2
Fuel Handling Accident Fission Product Inventory at 100 Hours Following Shutdown

Iodine	Isotopic Inventory, curies
I-131	2.99E5
I-132	2.53E5
I-133	3.15E4
I-134	0.000E0
I-135	2.25E1
Noble Gases	
Kr-85m	2.22E-2
Kr-85	4.72E3
Kr-87	0.00E0
Kr-88	0.00E0
Xe-131m	4.50E3
Xe-133m	1.06E3
Xe-133	5.75E5
Xe-135m	3.60E0
Xe-135	1.10E3
Xe-138	0.00E0

D.4 REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES

The calculated reactor coolant specific activities are included in Table D.4-1. The listed activities are maximum calculated values and were determined from the following assumptions:

1. Core thermal power (includes uncertainty) 1782.6 MWt
2. Operating time (full power) 550 days
3. Fraction of fuel rods containing clad defects 0.01
4. Reactor coolant liquid volume 5993 ft^3
5. Fission Product Escape Rates
   - Kr and Xe 6.5E-8 sec-1
   - Halogens, Rb and Cs 1.3E-8 sec-1
   - Y, Zr, Nb, Ru, Rh, La, Ce and Pr 1.6E-12 sec-1
   - Sr and Ba 1.0E-11 sec-1
   - Mo, Tc and Ag 2.0E-9 sec-1
   - Te 1.0E-9 sec-1
6. Normal purification flow rates
   - Mixed bed demineralizer 40 gpm
   - Cation bed demineralizer [1](effective) 4 gpm
7. Mixed bed demineralizer decontamination factors[2]
   - Kr and Xe 1.0
   - Br and I 10.0
   - Sr and Ba 10.0
   - Other 1.0
8. Cation bed demineralizer decontamination factor
   - Kr and Xe 1.0
   - Sr and BA 1.0
   - Rb-86, Cs-134 and Cs-137 10.0
   - Rb-88, Rb-89, Cs-136 and Cs-138 1.0
   - Other 1.0

1. Not in use during normal operations.
2. For all isotopes, except the isotopes of Kr, Xe, Br, I, Rb, Cs, Sr, and Ba, a removal decontamination factor (as distinct from demineralizer DF) of 10 was assumed. This is done to account for removal mechanisms other than ion exchange, such as plateout, etc.

9. Other removal mechanisms[3]
   (see footnote 3)
10. Volume control tank nominal volumes
    - Vapor volume 130 ft^3
    - Liquid volume 90 ft^3
11. Noble gas stripping fraction in unvented volume control tank
    - Kr-83m 7.9E-1
    - Kr-85 7.2E-5
    - Kr-85m 6.0E-1
    - Kr-87 8.4E-1
    - Kr-88 7.0E-1
    - Kr-89 9.9E-1
    - Xe-131m 1.8E-2
    - Xe-133 3.9E-2
    - Xe-133m 8.9E-2
    - Xe-135 3.6E-1
    - Xe-135m 9.5E-1
    - Xe-137 9.9E-1
    - Xe-138 9.6E-1

3. For all isotopes, except the isotopes of Kr, Xe, Br, I, Rb, Cs, Sr, and Ba, a removal decontamination factor (as distinct from demineralizer DF) of 10 was assumed. This is done to account for removal mechanisms other than ion exchange, such as plateout, etc.

Table D.4-1
RCS Coolant Fission and Corrosion Product Activities
(Based on 1% Fuel Defects)

Nuclide	Activity (μCi/gm)	Nuclide	Activity (μCi/gm)
Kr-83m	4.31E-01	Co-58	1.40E-02
Kr-85m	1.73E+00	Co-60	1.30E-03
Kr-85	8.60E+00	Sr-89	4.17E-03
Kr-87	1.13E+00	Sr-90	2.07E-04
Kr-88	3.28E+00	Sr-91	5.57E-03
Kr-89	9.260E-02	Sr-92	1.24E-03
Xe-131m	3.04E+00	Y-90	5.82E-05
Xe-133m	3.44E+00	Y-91m	3.01E-03
Xe-133	2.42E+02	Y-91	5.61E-04
Xe-135m	5.01E-01	Y-92	1.08E-03
Xe-135	8.69E+00	Y-93	3.58E-04
Xe-137	1.73E-01	Zr-95	6.53E-04
Xe-138	6.28E-01	Nb-95	6.55E-04
Br-83	9.11E-02	Mo-99	7.60E-01
Br-84	4.51E-02	Tc-99m	7.07E-01
Br-85	5.25E-03	Ru-103	5.59E-04
I-130	3.44E-02	Ru-106	1.89E-04
I-131	2.84E+00	Rh-103m	5.54E-04
I-132	2.89E+00	Rh-106	1.89E-04
I-133	4.24E+00	Ag-110m	1.54E-03
I-134	5.86E-01	Te-125m	6.92E-04
I-135	2.32E+00	Te-127m	3.13E-03
Rb-86	3.16E-02	Te-127	1.32E-02
Rb-88	4.09E+00	Te-129m	1.07E-03
Rb-89	1.87E-01	Te-131m	2.50E-02

Table D.4-1
RCS Coolant Fission and Corrosion Product Activities
(Based on 1% Fuel Defects)

Nuclide	Activity (μCi/gm)	Nuclide	Activity (μCi/gm)
Cs-134	2.86E+00	Te-131	1.35E-02
Cs-136	3.22E+00	Te-132	2.94E-01
Cs-137	2.16E+00	Te-134	2.87E-02
Ba-137m	2.04E+00	Ba-140	4.16E-03
Cs-138	9.63E-01	La-140	1.41E-03
H-3	3.50E+00	Ce-141	6.36E-04
Cr-51	5.40E-03	Ce-143	4.86E-04
Mn-54	4.00E-04	Ce-144	4.81E-04
Fe-55	2.10E-03	Pr-143	6.11E-04
Fe-59	5.10E-04	Pr-144	4.81E-04
Te-129	1.38E-02		

The above concentration are based on the following assumptions:

- Reactor coolant mass = 1.19×10^8 grams
- Fuel defects in fuel producing 1 percent of core power
- Reactor coolant letdown rate = 40 gpm
- No VCT purge

D.5 REACTOR COOLANT TRITIUM ACTIVITIES

During the fissioning of uranium, tritium atoms are generated in the fuel at a rate of approximately 8E-5 atoms per fission (1.05E-2 curies/Mwt-day). Other sources of tritium include neutron reactions with boron (in the coolant for shim control), neutron reactions with lithium (utilized in the coolant for pH control, and produced in the coolant neutron reactions with boron), and by neutron reactions with naturally occurring deuterium in light water.

D.5.1 Release Of Ternary-Produced Tritium

The tritium formed by ternary fission in uranium-fueled reactors, can be retained in the fuel, accumulate in the void between the fuel and cladding, react with cladding material (zirconium tritide), or diffuse through the cladding into the coolant. It is conservatively assumed that 10 percent of the ternary-produced tritium is released to coolant. This assumption then requires that the waste treatment system be sized to process approximately two reactor coolant system volumes in addition to normal reactor plant liquid wastes. Anticipated ternary tritium loss to the reactor coolant is 2 percent.

D.5.2 Tritium Produced from Boron Reactions

The neutron reactions with boron resulting in the production of tritium are:

- B-10 (n, 2α) T
- B-10 (n, α) Li-7 (n, n α) T
- B-11 (n, T) Be-9
- B-10 (n, d) *BE-9 (n, α) Li-6 (n, α) T

Of the above reactions, only the first two contribute significantly to the tritium production. The B-11 (n, T) Be-9 reaction has a threshold of 14 Mev and a cross section of ~ 5 Mb. Since the number of neutrons produced at this energy are less than 10^9 n/cm^2 sec the tritium produced from this reaction is negligible. The B-10 reaction may be neglected, since Be-9 has been found to be unstable.

D.5.3 Tritium Produced from Lithium Reactions

The neutron reactions with lithium resulting in the production of tritium are:

- Li-7 (n, nα) T
- Li-6 (n, α) T

In the Westinghouse-designed reactors, lithium is used to maintain the reactor coolant pH. The reactor coolant is maintained in accordance with EPRI Primary Water Guidelines. A chemical

volume cation bed demineralizer is included in the Chemical and Volume Control System to remove the excess lithium produced in the B-10 (n, α) Li-7 reactions.

The Li-6 (n, α) T reaction is controlled by limiting the Li-6 lithiating the demineralizers to <0.001 parts of Li-7. This limitation has been in effect on Westinghouse-designed reactors since 1962.

D.5.4 Tritium Produced from Deuterium Reactions

Since the amount of naturally occurring deuterium is less than 0.00015, the tritium produced from this reaction is less than 1 curie per year, and is therefore neglected.

D.5.5 Tritium Sources and Activities from the Reactor

Reactor coolant tritium sources and activities are given in Table D.5-1. The basic assumptions and plant parameters used in calculating these sources are:

1. Core thermal power — 1782.6 Mwt
2. Plant load factor — 0.9
3. Core water volume — 324.2 ft^3
4. Core water mass — 6.6E+06 g
5. Initial reactor coolant boron level
   - Initial cycle — 890 ppm
   - Equilibrium cycle — 1520 ppm
6. Reactor coolant volume — 5993 ft^3
7. Reactor coolant peak lithium level (99 percent pure Li^7) — 3.5 ppm
8. Fraction of ternary tritium diffusing through zirconium cladding
   - Design value — 0.10
   - Expected value — 0.02

NOTE: Based on considering tritium release data from Westinghouse 4-loop plants that have had limited use of stainless steel clad burnable poison rods and have a long operating history, the recommended “best estimate” release fraction is 0.02 or 2 percent for components clad with zirconium-based alloys. The value is applied to the amount of tritium produced within the component over a relatively long-term, i.e., a fuel cycle. (Note that it is defined as the fractional release of that produced and not the existing inventory). This value is consistent with data from the Trojan and Prairie Island units, which are considered to be the best candidate plants relative to identification of the best estimate core release fraction; i.e., no extended cycle operation and mirror impact of burnable poison addition.

Table D.5-1
Reactor Coolant Tritium Activities (curies per cycle)

Tritium Source	Total Produced	Released to the Coolant	
		Design Value	Expected Value
Ternary Fissions	8750	875	175
Soluble Poison Boron	355	355	355
Li-7 Reaction	13.1	13.1	13.1
Li-6 Reaction	89.5	89.5	89.5
Deuterium Reaction	1.7	1.7	1.7
Total – Equilibrium Cycle	9209	1334	634

D.6 VOLUME CONTROL TANK INVENTORY

The 220-cubic foot volume control tank is assumed to contain 90 cubic feet of liquid and 130 cubic feet of vapor. Table D.6-1 lists the fission product inventory in the volume control tank with clad defects in 1 percent of the fuel rods. The assumptions used in calculating these activities are the same as those given in Section D.4.

Table D.6-1
Volume Control Tank Inventory

Nuclide	VCT Inventory (Ci)
Kr-85m	6.29E1
Kr-85	7.35E2
Kr-87	1.64E1
Kr-88	8.85E1
Xe-131m	2.07E2
Xe-133m	2.21E2
Xe-133	1.62E4
Xe-135m	2.79E1
Xe-135	4.52E2
Xe-138	1.94E0
I-131	8.69E-1
I-132	8.85E-1
I-133	1.30E0
I-134	1.79E-1
I-135	7.09E-1

D.7 GAS DECAY TANK INVENTORY

Radiological inventories for one gas decay tank are determined by assuming that the Reactor Coolant System noble gas activities presented in Table D.4-1 are available for degassing of the Reactor coolant System at the end of cycle. It is assumed that no purging of the Volume Control Tank occurs during the cycle. The inventory is calculated assuming nuclide decay, degassing of the reactor coolant with letdown at the maximum rate, and periodic purging to the gas decay tank. The maximum inventory for each nuclide during the degas and purge cycle is selected for the inventory. The resulting values are given in Table D.7-1.

Table D.7-1
Inventory of Gas Decay Tank After Shutdown and Degassing of the RCS
(Based on 1 percent of Fuel Defects)

Isotope	GDT_Inventory (Ci)
Kr-85m	8.53E1
Kr-85	2.39E3
Kr-87	1.58E1
Kr-88	1.08E2
Xe-131m	5.20E2
Xe-133m	4.76E2
Xe-133	3.85E4
Xe-135m	2.78E1
Xe-135	6.68E2
Xe-138	1.84E0

D.8 DOSE CALCULATION MODELS FOR SAFEGUARDS ANALYSIS

D.8.1 Introduction

This section identifies the models used to calculate the offsite radiation dose that would result from radioactivity release due to various postulated accidents. The postulated accidents are:

- Fuel handling accident
- Volume control tank rupture
- Gas decay tank rupture
- Rupture of a steam generator tube
- Rupture of a main steam line
- RCCA Ejection
- Loss of coolant accident

D.8.2 Assumptions

The total effective dose equivalent (TEDE) doses are determined at the site boundary for the limiting two hour interval, at the low population zone (LPZ) and to control room personnel (CR) for the duration of the event. The dose conversion factors (DCFs) used in determining the committed effective dose equivalent (CEDE) or inhalation dose are from Reference 1. The TEDE dose is equivalent to the CEDE dose plus the acute dose for the duration of exposure to the cloud. The CEDE dose DCFs are given in Table D.8-1.

The γ-body (acute) doses are based on the noble gas dose conversion factors from International Commission on Radiological Protection (ICRP) Publication 30 (Reference 3) and based on average disintegration energies for the remainder of the nuclides. The dose conversion factors and average disintegration energies for the γ-body doses are listed in Table D.8-1.

The offsite breathing rates and the offsite atmospheric dispersion factors used in the offsite radiological calculations are provided in Table D.8-3 and Table D.8-4.

Parameters modeled in the control room personnel dose calculations are provided in Table D.8-5. These parameters include the normal operation flow rates, the emergency operation flow rates, control room volume, filter efficiencies and control room operator breathing rates. The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for release of activity from the release point to the control room intake are used to determine the activity available at the intake. The inflow (filtered and unfiltered) to the control room and the control room recirculation flow are used to calculate the activity introduced to the control room and cleanup of activity from that flow.

The following assumptions are basic to the models used:

1. Direct radiation from the source point is negligible compared to whole-body radiation due to submersion in the radioactive leakage cloud.
2. All radioactive material releases are treated as ground level sources regardless of the point of discharge. The atmospheric dispersion factors discussed in Section D.8.2 apply to all radioactive releases.
3. The dose receptor is a standard man as defined by the ICRP (Reference 5).
4. Radioactive decay from the point of release to the dose receptor is neglected.

D.8.3 Dose Calculation Models

Offsite Dose Calculation Models

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

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

where:

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

DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci) (Table D.8-1)

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

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table D.8-3)

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table D.8-4)

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

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

where:

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

DCF_i = gamma dose conversion factor via external exposure for isotope i (rem-m^3/Ci-sec) (Table D.8-1)

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

$(\chi/Q)_j$ = atmospheric dispersion factor during time interval j (sec/m^3) (Table D.8-4)

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

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

where:

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

$\bar{E}\,\gamma_i$ = gamma disintegration energy for isotope i (mev/dis) (Table D.8-1)

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

$(\chi/Q)_j$ = satmospheric dispersion factor during time interval j (sec/m^3) (Table D.8-4)

Control Room Dose Calculation Models

Reference 6 demonstrates that the 30 day dose to control room operators is within the limit specified in 10 CF 50.67 for all events.

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

The control room is modeled as a discrete volume. The atmospheric dispersion factors calculated for the transfer of activity to the control room intake are used to determine the activity available at the control room intake. The inflow (filtered and unfiltered) to the control room and filtered recirculation flow are used to calculate the concentration of activity in the control room. Control room parameters used in the analyses are presented in Table D.8-5.

Control room inhalation doses are calculated using the following equation:

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

where:

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

DCF_i = CEDE dose conversion factor via inhalation for isotope i (rem/Ci) (Table D.8-1)

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

$(BR)_j$ = breathing rate during time interval j (m^3/sec) (Table D.8-5)

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

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

where:

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

GF = geometry factor, calculated based on Reference 4, using the equation:

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

DCF_i = gamma dose conversion factor via external exposure for isotope i (rem • m^3/Ci-sec) (Table D.8-1)

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

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

$$D_\gamma = 0.25\left(\frac{1}{GF}\right) * \sum_i \overline{E}\gamma_i \left(\sum_j Conc_{ij}\right)$$

where:

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

GF = geometry factor, calculated based on Reference 4, using the equation:

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

$\overline{E}\gamma_i$ = gamma disintegration energy for isotope i (mev/dis) (Table D.8-1)

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

Table D.8-1
Nuclide Parameters

Nuclide	Decay Constant (hr^{-1})	CEDE DCF (rem/Ci inhaled)	Average Gamma Disintegration Energies (Mev/dis)	Noble Gas DCF (rem m^3/Ci sec)
I-131	0.00359	3.29E4	0.38	NA
I-132	0.303	3.81E2	2.2	NA
I-133	0.0333	5.85E3	0.6	NA
I-134	0.791	1.31E2	2.6	NA
I-135	0.105	1.23E3	1.4	NA
Kr-85m	0.155	NA	NA	0.0307
Kr-85	7.37E-6	NA	NA	0.000484
Kr-87	0.547	NA	NA	0.146
Kr-88	0.248	NA	NA	0.37
Xe-131m	0.00241	NA	NA	0.00152
Xe-133m	0.0130	NA	NA	0.00553
Xe-133	0.00546	NA	NA	0.00624
Xe-135m	2.72	NA	NA	0.0775
Xe-135	0.0756	NA	NA	0.0482
Xe-138	2.93	NA	NA	0.198
Cs-134	3.84E-5	4.62E4	1.55	NA
Cs-136	2.2E-3	7.33E3	2.16	NA
Cs-137	2.64E-6	3.19E4	0.564	NA
Rb-86	1.55E-3	6.63E3	0.0945	NA
Te-127m	2.65E-4	2.15E4	0.0112	NA
Te-127	7.41E-2	3.18E2	4.86E-3	NA
Te-129m	8.6E-4	2.39E4	0.0375	NA
Te-129	0.598	9.0E1	0.0591	NA
Te-131m	2.31E-2	6.4E3	1.42	NA

Table D.8-1
Nuclide Parameters

Nuclide	Decay Constant (hr^{-1})	CEDE DCF (rem/Ci inhaled)	Average Gamma Disintegration Energies (Mev/dis)	Noble Gas DCF (rem m^3/Ci sec)
Te-132	8.86E-3	9.44E3	0.233	NA
Sb-127	7.5E-3	6.04E3	0.688	NA
Sb-129	0.16	6.44E2	1.44	NA
Sr-89	5.72E-4	4.14E4	8.45E-5	NA
Sr-90	2.72E-6	1.3E6	0.0	NA
Sr-91	0.073	1.66E3	0.693	NA
Sr-92	0.256	8.1E2	1.34	NA
Ba-139	0.502	1.7E2	0.043	NA
Ba-140	2.27E-3	3.74E3	0.182	NA
Ru-103	7.35E-4	8.95E3	0.468	NA
Ru-105	0.156	4.55E2	0.775	NA
Ru-106	7.84E-5	4.77E5	0.0	NA
Rh-105	1.96E-2	9.56E2	0.078	NA
Mo-99	1.05E-2	3.96E3	0.15	NA
Tc-99m	0.115	3.3E1	0.126	NA
Ce-141	8.89E-4	8.96E3	0.076	NA
Ce-143	0.021	3.39E3	0.282	NA
Ce-144	1.02E-4	3.74E5	0.021	NA
Pu-238	9.02E-7	3.92E8	1.81E-3	NA
Pu-239	3.29E-9	4.3E8	8.08E-4	NA
Pu-240	1.21E-8	4.3E8	1.73E-3	NA
Pu-241	5.5E-6	8.26E6	2.54E-6	NA
Np-239	0.0123	2.51E3	0.172	NA
Y-90	1.08E-2	8.44E3	1.7E-6	NA

Table D.8-1
Nuclide Parameters

Nuclide	Decay Constant (hr^{-1})	CEDE DCF (rem/Ci inhaled)	Average Gamma Disintegration Energies (Mev/dis)	Noble Gas DCF (rem m^3/Ci sec)
Y-91	4.94E-4	4.89E4	3.61E-3	NA
Y-92	0.196	7.80E2	0.251	NA
Y-93	0.0686	2.15E3	0.0889	NA
Nb-95	8.22E-4	5.81E3	0.766	NA
Zr-95	4.51E-4	2.37E4	0.739	NA
Zr-97	4.1E-2	4.33E3	0.179	NA
La-140	1.72E-2	4.85E3	2.31	NA
La-141	0.176	5.81E2	0.0427	NA
La-142	0.45	2.53E2	2.68	NA
Nd-147	2.63E-3	6.85E3	0.14	NA
Pr-143	2.13E-3	1.09E4	8.9E-9	NA
Am-241	1.83E-7	4.44E8	0.0324	NA
Cm-242	1.77E-4	1.73E7	1.83E-3	NA
Cm-244	4.37E-6	2.48E8	1.7E-3	NA

Notes: CEDE = Committed effective dose equivalent
DCF = Dose conversion factor

Table D.8-3
Standard Man Breathing Rates

Time Period	Breathing Rate, m^3/sec
0–8 hours	3.47E-4
8–24 hours	1.75E-4
> 24 hours	2.32E-4

Table D.8-4
Off-Site Atmospheric Dispersion Factors

Location and Time Period	Atmospheric Dispersion Factor sec/m^3
Site Boundary (1200 ft)	2.232E-4
Low Population Zone	
0–2 hours	3.977E-5
2–24 hours	4.100E-6
1–2 days	2.427E-6
> 2 days	4.473E-7

Table D.8-5
Control Room Parameters

Breathing Rate – Duration of the Event	3.47E-4 m^3/sec
Control Room Volume	127,600 ft^3
Atmospheric Dispersion Factors	
0–8 hours	2.93E-3 sec/m^3
8–24 hours	1.73E-3 sec/m^3
1–4 days	6.74E-4 sec/m^3
4–30 days	1.93E-4 sec/m^3
Occupancy Factors[a]	
0–24 hours	1.0
1–4 days	0.6
4–30 days	0.4
Normal Ventilation Flow Rates	
Filtered Makeup Flow Rate	0.0 scfm
Filtered Recirculation Flow Rate	0.0 scfm
Unfiltered Makeup Flow Rate	2500 scfm (±10%)
Unfiltered Recirculation Flow Rate	(Not modeled – no impact on analyses)
Emergency Ventilation System Flow Rates	
Filtered Makeup Air Flow Rate	0.0 scfm
Filtered Recirculation Flow Rate	2500 scfm (±10%)
Unfiltered In-leakage	200 scfm
Filter Efficiencies	
Elemental	90%
Organic	90%

a. These occupancy factors have been conservatively incorporated in the atmospheric dispersion factors. This is conservative, since it does not allow the benefit of reduced occupancy for activity already present in the Control Room from earlier periods.

Table D.8-5
Control Room Parameters

Filter Efficiencies (continued)	
Particulate	99%
R-23 Sensitivity	2.32E7 (cpm/μCi/cc for Xe-133)
Setpoint for Control Room Isolation	1.0E4 (cpm)
Location of R-23	R-23 is located at a junction of the intake and the recirculation ducts such that it monitors the mixed air stream
Delay to initiate switchover of HVAC from normal operation to emergency operation after SI signal	63 seconds
Delay for switchover of HVAC from normal operation to emergency operation after receipt of an isolation signal	<10 seconds

REFERENCES - APPENDIX D

1. 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
2. CCC-371, “ORIGEN2.1: Isotope Generation and Depletion Code – Matrix Exponential Method,” RSIC Computer Code Collection, Oak Ridge National Laboratory, February 1996
3. International Commission on Radiological Protection, “Limits for Intakes of Radionuclides by Workers,” ICRP Publication 30, 1979
4. Murphy, K. G., Campe, K. M., “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19,” Proceedings of the Thirteenth AEC Air Cleaning Conference held August 1974, published March 1975
5. Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959), *Health Physics Vol. 3*, p. 30, 146-153, 1960
6. John Lamb (NRC) to Tom Coutu (NMC) transmitting the NRC SER For Amendment No. 166 to the Operating License, approving Implementation of Alternate Source Term, Letter No. K-03-040, March 17, 2003