CHAPTER 11 – RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

This section presents information on the sources of radioactivity that serve as design bases for the various radioactive waste treatment systems for normal operation, including anticipated operational occurrences (expected source term), as well as for design conditions (design basis source term). The application of the source terms and the mathematic models and parameters used to calculate source terms for normal operation and for design conditions are different. A clear distinction is made between the design basis source term and the expected source term. The shielding source term for the radiation shield design is addressed in Section 12.2, and the accident source term used for the radiological accident evaluation is addressed in Chapter 15.

Definitions

a. Design basis source term

The design basis source term is used for the design of the radioactive waste management system and for determining design lifetime integrated doses for the design specifications of plant equipment. The design basis source term is based on design basis data for the maximum reactor coolant activity as shown in Table 11.1-1.

b. Expected source term

The expected or operating basis source term is used for describing annual releases from the plant to the environment on an average basis. Site boundary doses due to releases from the plant ventilation exhausts, liquid discharges, and offsite shipment of solid radioactive material are examples of calculations that use this source term. The expected source term is based on a realistic model as described in ANSI/ANS 18.1 (Reference 1) for reactor coolant activity during normal operation as represented in Table 11.1-1. Calculations pertaining to releases described in 10 CFR 50, Appendix I (Reference 2), conform to the methods and parameters described in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.112 (Reference 3).

11.1.1 Design Basis Source Term

11.1.1.1 Fission Product Activities in the Reactor Coolant

The DAMSAM Code (Reference 4) is used to calculate the design basis source term of fission products in the reactor coolant for the design of the radioactive waste management system and to determine design lifetime integrated doses for plant equipment. The isotopes considered in the maximum case are those that are significant for design purposes by reason of a combination of energy, half-life, and/or abundance.

The mathematical model used to determine the concentration of nuclides in the reactor coolant system (RCS) involves a group of linear, first-order differential equations. These equations are obtained by applying a mass balance for production and removal in both the fuel pellet region and the reactor coolant region.

In the fuel pellet region, the mass balance includes fission product production by direct fission yield, by parent fission product decay, and by neutron activation, while removal includes decay, neutron activation, and escape to the reactor coolant.

In the reactor coolant region, the fission product is introduced when it escapes from the fuel pellet through defective fuel rod cladding, parent decay in the reactor coolant, and neutron activation of the fission product in the reactor coolant. The fission product is removed by decay; coolant purification; boron feed-and-bleed operations (to accommodate fuel burnup); leakage and other feed-and-bleed operations during startups, shutdowns, and load-following operation; and neutron activation.

The expression to determine the fission product inventory in the fuel pellet region is:

$$\frac{dN_{c,i}}{dt} = (F)(Y_i)(P) + (f_{i-1}\lambda_{i-1})N_{p,i-1} + \sigma_j\phi N_{p,j} - (\lambda_i + Dv_i + \sigma_i\phi)N_{p,i} \qquad \text{(Eq. 11.1-1)}$$

The expression to determine the fission product inventory in the reactor coolant region is:

$$\frac{dN_{c,i}}{dt} = (D)(v_i)\left(N_{p,i}\right) + (f_{i-1}\lambda_{i-1})N_{c,i-1} + (\sigma_j\phi CVR)\,N_{c,j}$$
$$-\left({}^{\dot{Q}}\lambda_i + \frac{\dot{Q}}{w}\eta_i + \frac{(1-\eta_i)\dot{C}}{C_0 - t\dot{C}} + \frac{L}{W} + \sigma_i\phi CVR\right) N_{c,i} \qquad \text{(Eq. 11.1-2)}$$

Where the variables are defined as:

N	=	nuclide population, atoms
F	=	average fission rate, fissions/MWt-sec
Y	=	core averaged fission yield of nuclide, fraction
P	=	core power, MWt
λ	=	decay constant, sec^{-1}
σ	=	microscopic capture cross section, cm^2
ϕ	=	thermal neutron flux, neutrons/cm^2-sec
v	=	escape rate coefficient, sec^{-1}
f	=	branching fraction
t	=	time, seconds
D	=	defective fuel cladding, fraction
CVR	=	ratio of core coolant volume to reactor coolant volume
$\dot{Q}$	=	chemical and volume control system (CVCS) purification mass flow rate during power operation, kg/sec
W	=	RCS mass during power operation, kg
η	=	resin efficiency of CVCS ion exchanger and gas stripper efficiency
C_0	=	boron concentration at the beginning of core life, ppm
$\dot{C}$	=	boron concentration reduction rate due to feed and bleed, ppm/sec
L	=	mass flow rate of reactor coolant leakage or the other feed and bleed, kg/sec

Where the subscripts are identified as:

i	=	i^{th} nuclide
i-1	=	precursor to i^{th} nuclide for decay

i-1	=	precursor to i^{th} nuclide for decay
j	=	j^{th} nuclide to i^{th} nuclide for neutron activation
p	=	pellet region
c	=	reactor coolant region

This model does not involve the fuel plenum and gap region. Instead, escape rate coefficients are used to represent the overall release from the fuel pellets to the reactor coolant. The escape rate coefficient is an empirical value derived from experiments involving the nuclear research reactor (NRX) and material testing reactor (MTR) that were initiated at the Bettis Plant (Reference 5). The escape rate coefficients were obtained from test rods that were operated at high linear heat rates.

The linear heat rates were uniform over test sections of 26.035 cm (10.25 in) in length. The exact linear heat rates were not known, but post-irradiation inspection showed that some test specimens had experienced centerline melting. Supplemental tests were conducted in Canada to determine the effect of rod length on the release of fission gases and iodines from the defective fuel rods (Reference 6). The experiments also determined the relationship between linear heat rate and the escape rate coefficient. Because the average heat rate for a fuel rod is below the linear heat rate of 591 W/cm (18 kW/ft), which corresponds to the selected escape rate coefficients for halogens and noble gases shown in Table 11.1-1, the current escape rate coefficients are conservative.

Table 11.1-1 shows the values of the parameters that are used to calculate the reactor coolant fission product source term. The maximum reactor coolant fission product source term used for the design is presented in Table 11.1-2.

The design basis reactor coolant activities are based on 1 percent fuel cladding defects. The total activities of iodine and noble gases are 3.6 μCi/g (I-131 dose equivalent) and 580 μCi/g (Xe-133 dose equivalent), respectively. The activities of iodine and noble gases are limited to 1.0 μCi/g (I-131 dose equivalent) and 300 μCi/g (Xe-133 dose equivalent) by the plant technical specification. The reactor coolant activities that are limited by the Technical Specification during normal power operation are lower than the design values.

11.1.1.2 Spent Fuel Pool and Refueling Pool Activities

Table 11.1-3 contains the assumptions used in calculating the design basis specific activities of the fission and corrosion products in the spent fuel pool for the start of the refueling period, and Table 11.1-4 contains the results of the calculations. The RCS is assumed to cool down for 2 days upon shutdown for refueling. During this period, the primary coolant is let down through the purification filter, purification ion exchanger, gas stripper, and volume control tank. The letdown serves two purposes: (1) removing the noble gases in the gas stripper prevents large activity releases from the refueling pool to the reactor containment building following reactor vessel head removal and (2) reducing, through ion exchange and filtration, the dissolved fission and corrosion products in the reactor coolant that would otherwise enter the spent fuel pool and refueling pool.

At the end of this period, the reactor coolant above the reactor vessel flange is partially drained. The reactor vessel head is unbolted, and the refueling pool is filled with water from the in-containment refueling water storage tank (IRWST). The remaining reactor coolant containing radioactivity is then mixed with water in the refueling pool and spent fuel pool. Refueling pool water is cooled by the shutdown cooling system and cleaned of radioactivity by the spent fuel pool cooling and cleanup systems. The spent fuel pool water is cooled and cleaned of radioactivity by the spent fuel pool cooling and cleanup systems.

After refueling, the spent fuel pool is isolated, and the water in the refueling pool is returned to IRWST. The total activity in the spent fuel pool is determined through these serial processes.

Leakage of radionuclides into the spent fuel pool from damaged fuel stored in the pool is not considered a significant contributor to the radionuclide concentration in the spent fuel pool water because of the extremely low escape rate coefficients of the spent fuel in the spent fuel pool. The low escape rate coefficients are due in part to the low spent fuel pool temperature. Most of the activity releases from the defective fuel elements during shutdown and cooldown of the reactor prior to removal of the reactor vessel head. If significant releases from the defective fuel are detected, the defective fuel elements are isolated in a separate container so the released activity does not contribute to the specific activity in the spent fuel pool water. The primary source of radioactivity in the spent fuel pool water, after refueling operations have been completed, is due to displaced activation products, which is crud, from the surfaces of the spent fuel assemblies.

11.1.1.3 Secondary System Activity

For the purpose of the design, the steam generator is assumed to have tube leaks, and radionuclides are moved to the secondary system from the primary system. In determining design basis sources in the secondary system, a total steam generator tube leakage is assumed to be 3,270 L/day (0.6 gal/min), which is the accident-induced steam generator leakage criterion (Chapter 16, Subsection 5.5.9.b.2). This value is conservatively used even though the limiting condition of operation for steam generator operational leakage in the Technical Specifications is 562 L/day (0.1 gal/min) (Chapter 16, Subsection 3.4.12).

Radionuclides are removed from the secondary system by the following mechanisms:

a. Steam generator blowdown demineralizer treatment

b. Condensate polishing demineralizer treatment

c. Radioactive decay

d. Exhaust through the main condenser vacuum pumps

e. Main steam leakage

Primary coolant activities used to determine the design basis sources in the secondary system are addressed in Subsection 11.1.1.1. Assumptions used in determining the secondary system activities are listed in Table 11.1-5. Design basis equilibrium radionuclide concentrations in the secondary system are determined as described below:

a. Steam generator liquid activity

   The following expression determines the concentration of nuclides in the steam generator liquid (i.e., blowdown source):

$$M_{sl}\frac{dN_{sl}}{dt} = RN_w - TN_{sg} - BN_{sl} - \lambda N_{sl}M_{sl} + TN_{sg}(1-F)$$

$$+TFN_{sg}\left[\frac{0.8333}{DF_d} + \frac{0.1667}{DF_dDF_c}\right] + BN_{sl}\frac{1}{DF_b}\left[0.8333 + \frac{0.1667}{DF_c}\right] \qquad \text{(Eq. 11.1-3)}$$

Where the variables are defined as:

N = nuclide concentration, Bq/g

R = primary-to-secondary leakage rate, g/sec

T = main steam flow rate, g/sec

F = fraction of radionuclide in the main steam reaching to the main condenser

M = secondary liquid mass in steam generator, g

B = steam generator blowdown rate, g/sec

DF = decontamination factor

λ = decay constant, sec^{-1}

0.1667 = fraction of condensate water processed in condensate polishing system (0.8333 means fraction of bypassing the condensate polishing system)

Where the subscripts are defined as:

s = steam generator

w = RCS

c = condensate polishing system

d = condenser vacuum system

b = steam generator blowdown system

l = liquid in the secondary system

g = steam in the secondary system

Also, $N_{sg} = \alpha N_{sl}$

Where:

α = steam generator partition coefficient (the ratio of concentration-in-steam to concentration-in-liquid)

Therefore, the equilibrium concentration of nuclides in the steam generator liquid is given by:

$$N_{sl} = \frac{RN_w}{\lambda M_{sl} + \alpha TF\left\{1 - \left(\frac{0.8333}{DF_d} + \frac{0.1667}{DF_d DF_c}\right)\right\} + B\left\{1 - \frac{1}{DF_b}\left(0.8333 + \frac{0.1667}{DF_c}\right)\right\}} \quad \text{(Eq. 11.1-4)}$$

Parameter definition and units are shown in Eq. 11.1-3.

The design basis radionuclide concentrations in the steam generator liquid are listed in Table 11.1-6.

a. Main steam activity

To obtain the peak noble gas concentration in the main steam leaving the steam generators, all of the noble gases that enter the steam generator via primary coolant leakage are assumed to exit with the steam (i.e., no noble gases are in the steam generator secondary liquid). The noble gas activities in the steam exiting the steam generator are the ratio of the primary-to-secondary (PTS) leakage rate multiplied by the radionuclide concentration in the primary coolant to the steam flow rate out of the steam generator.

This is expressed as:

$$N_{sg} = \frac{R \cdot N_W}{T} \quad \text{(Eq. 11.1-5)}$$

The parameter definition and units are shown in Eq. 11.1-3.

The equilibrium concentration of non-noble gas radionuclides in the steam exiting the steam generator is the equilibrium concentration of the radionuclides in the steam generator liquid multiplied by the steam generator partition coefficient, which is the ratio of concentration in the steam generator steam to the concentration in the steam generator water.

This is expressed as:

$$N_{sg} = \alpha \cdot N_{sl} \qquad \text{(Eq. 11.1-6)}$$

Parameter definition and units are shown in Eq. 11.1-3.

The design basis radionuclide concentrations in the main steam are determined using Equations 11.1-5 and 11.1-6 and are listed in Table 11.1-6.

b. High-capacity blowdown liquid activity

Approximately once a week, a high-capacity blowdown is performed to remove accumulated crud in the steam generator. The high-capacity blowdown is performed for 2 minutes per week. For the two steam generators, the high-capacity blowdown rate is 5 percent of the NSSS maximum steaming rate (113 kg/sec [249.0 lb/sec]) based on cold leg temperature, and the minimum value is 3.6 percent (81.2 kg/sec [179.2 lb/sec]) based on hot leg temperature. For conservatism, the flow rate of 81.2 kg/sec (179.2 lb/sec) is used to calculate radionuclide crud concentration in the high-capacity blowdown liquid. To obtain the radionuclide crud activity in the high-capacity blowdown liquid, it is assumed that the crud radionuclides (i.e., Mn, Co, Fe, Cr, Zr) due to PTS coolant leakage remain in the steam generators between high-capacity blowdown operations. The accumulated radionuclide crud is diluted with 9.78×10^3 kg (2.15×10^4 lb) of the high-capacity blowdown water and discharged to the high-capacity blowdown flash tank. The radionuclide crud concentrations in the high-capacity blowdown liquid are calculated as follows:

$$N_h = \frac{R \cdot N_W}{\lambda \cdot M_h}\{1 - e^{-xt}(-\lambda t)\} \qquad \text{(Eq. 11.1-7)}$$

Where:

N_h = radionuclide concentration of crud within the high-capacity blowdown water, Bq/g

t = period of high-capacity blowdown, sec

λ = decay constant, sec^{-1}

M_h = mass of high-capacity blowdown water, g

R and N_w are described in the variable identifications for Eq. 11.1-3.

The remaining radionuclide concentrations in the high-capacity blowdown liquid (i.e., non-noble gas and non-crud radionuclides) are determined using Eq. 11.1-4, which determines the equilibrium radionuclide concentration in the steam generator liquid.

Radionuclide crud concentrations in the high-capacity blowdown liquid are listed in Table 11.1-7.

11.1.1.4 Radwaste System Activities

Source terms for the liquid waste management system (LWMS) are described in Section 11.2. Source terms for the gaseous waste management system (GWMS) are described in Section 11.3.

11.1.1.5 Volume Control Tank Activity

The total activity inventory in the volume control tank (VCT) is based on an expected maximum water volume, 15,725 L (4,154 gal), of reactor coolant letdown and an expected maximum vapor volume of 17,500 L (618 ft^3). The design basis specific activities of gaseous sources vented from the VCT to the gaseous waste management system (GWMS) are provided in Table 11.1-8.

11.1.1.6 Reactor Drain Tank Activity

The total activity inventory in the reactor drain tank (RDT) is based on an expected maximum water volume of 11,962 L (3,160 gal) and an expected maximum vapor volume of 9,085 L (321 ft^3). The design basis specific activities of gaseous sources vented from the RDT to the GWMS are provided in Table 11.1-8.

11.1.1.7 Gas Stripper Activity

The total activity inventory in the gas stripper is based on the summation of activity in the aftercooler, the heat recovery exchanger, the overhead condenser, the reboiler, the stripper column, and the feed preheater within the gas stripper package. The design basis specific activities of gaseous sources vented from the gas stripper to the GWMS are provided in Table 11.1-8.

11.1.1.8 Equipment Drain Tank Activity

The total activity inventory in the equipment drain tank (EDT) is based on an expected maximum water volume of 13,306 L (3,515 gal) and an expected maximum vapor volume of 33,085 L (1,168 ft^3). The design basis specific activities of gaseous sources vented from the EDT to the GWMS are provided in Table 11.1-8.

11.1.2 Expected Source Term

11.1.2.1 Reactor Coolant Activities

The data in Table 11.1-9 represent the expected normal fission and corrosion product specific activities in the reactor coolant with no gas stripping. The data are used in evaluating only normal operations including anticipated operational occurrences. The expected specific activities in the reactor coolant are based on ANSI/ANS 18.1 using the normal operating parameters provided in Table 11.1-1.

11.1.2.2 Spent Fuel Pool and Refueling Pool Activities

The model used to determine the spent fuel pool and refueling pool radionuclide activities is described in Subsection 11.1.1.2. The model used to predict expected activities is the same as the analysis model of the design basis source term except that the expected source terms in the primary coolant are used. The expected specific activities for the spent fuel pool and refueling pool are shown in Table 11.1-4.

11.1.2.3 Secondary System Activities

The equilibrium radionuclide concentrations in the steam generator liquid and in the main steam during the normal operation are determined using the method described in Subsection 11.1.1.3. The steam generator tube leak rate from the primary to the secondary system is assumed to be 34 kg/day (75 lb/day), based on ANSI/ANS 18.1. Additional assumptions used to determine the secondary activity are provided in Table 11.1-5.

The expected specific activities for the secondary system are provided in Table 11.1-10.

11.1.3 Neutron Activation Products

11.1.3.1 Deposited Crud Activities

Deposited crud activities on primary system surfaces have been evaluated using measured data from various operating pressurized water reactors. Even though these reactors have different water chemistries and different materials in contact with the primary coolant, their crud activities (Bq/g-crud), crud film thicknesses, and dose rates are remarkably similar. The half-lives, reactions, and gamma decay energies for each of the long-lived isotopes in the radioactive crud are as provided in Table 11.1-11.

The radioactive crud originates from in-core and out-of-core surfaces. The radioactive crud deposits on the in-core surfaces and erodes after a short irradiation period. This irradiation period or core residence time (t_{res}) for each isotope is determined by the following equations see Appendix 11A for the derivation of these equations:

Circulating crud:

$$t_{res} = \frac{1}{\lambda_i} \ln\left(1 - \frac{A_i A_T}{\Sigma_i \phi A_c}\right) \qquad \text{(Eq. 11.1-8)}$$

Deposited crud:

$$t_{res} = \frac{1}{\lambda_j} \ln\left(1 - \frac{A_j}{\Sigma_j \phi}\right) \qquad \text{(Eq. 11.1-9)}$$

Where:

A_i, A_j = crud activities for each isotope, Bq/g-crud

$\Sigma_i\phi$, $\Sigma_j\phi$ = activation rate for each isotope, reaction/g-sec

A_T = total primary system area, cm^2

A_C = core surface area, cm^2

λ_i, λ_j = decay constant for each isotope, sec^{-1}

t_{res} = core residence time, sec

The activation cross section (Σ_i) is:

$$\Sigma_i = \frac{(a/o)_i (w/o)_i N_o \sigma_i}{\{A\}_i} \qquad \text{(Eq. 11.1-10)}$$

Where:

$(a/o)_i$ = isotopic abundance, fraction

$(w/o)_i$ = elemental abundance in the crud or the elemental abundance in the base metal, fraction

N_o = Avogadro's number, 6.023×10^{23} atoms/g-mole

$\{A\}_i$ = atomic weight of isotope (i)

σ_i = microscopic cross section, cm^2

Σ_i = activation cross section, cm^2/g

The core residence times are determined by applying the measured average and maximum crud activities (Bq/g-crud) from various operating reactors, system parameters, and activation rates to the above expressions. The core residence times are shown in Table 11.1-12. The crud activities (A_i) are determined by applying the averages (t_{res}) of the maximum core residence times in Table 11.1-12, the system parameters, and the activation rates to the following equation. Because all of the Fe-59 residence times are long, the activity (A_j) is assumed to be saturated.

$$A_i = \Sigma_i \left(1 - e^{-\lambda_i t_{res}}\right) \frac{A_C}{A_T} \qquad \text{(Eq. 11.1-11)}$$

Where:

A_i = crud activities, Bq/g-crud

As the averages (t_{res}) of the maximum core residence times are, in general, a factor of 2 to 4 greater than a straight average residence time, the resulting calculated crud activities are conservative. These calculated crud activities of the long-lived isotopes are as shown in Table 11.1-13. These calculated crud activities are applied to both the circulating crud and out-of-core deposited crud.

Applying the average crud level (0.075 ppm) in the reactor coolant of various operating reactors to the calculated crud activities (Bq/g-crud) in Table 11.1-13, the crud specific activities in the reactor coolant, as shown in Table 11.1-14, are determined. The partial crud activities in the reactor coolant from the above conservative evaluation can be less than the expected crud activities (Table 11.1-9) in the reactor coolant during normal operations. In this case, the expected crud activities during normal operations are used as design basis maximum activities. These circulating crud activities in the reactor coolant are listed in Table 11.1-2.

The maximum coolant activities can be greater due to "crud bursts" during shutdown or changes in reactor power level. However, these "bursts" occur over short periods, and the average values are therefore more reasonable for use during long-term operation.

11.1.3.2 Carbon-14 Production

Carbon-14 is produced by neutron activation of O^{17} and N^{14} isotopes in the reactor coolant system (RCS). The greatest amount of C-14 is produced by the O^{17} (n, α) C^{14}reaction, and a lower amount of C-14 is produced by the N^{14}(n, p)C^{14} reaction. The production rate of C-14 (Q, Bq/cycle) from both reactions can be calculated by using the following equation:

$$Q = \lambda\ t\ m\ N\ (\sigma_{th}\phi_{th} + \sigma_f\ \phi_f) \qquad \text{(Eq. 11.1-12)}$$

Where:

λ = decay constant, 3.84×10^{-12} sec^{-1}

t = reactor operating time, 4.15×10^{7} sec

m = mass of active core water, 1.64×10^7 g

N = atom concentration in the RCS water
{$N(O^{17}) = 1.27 \times 10^{19}$ atoms/g H_2O, $N(N^{14})=1.31 \times 10^{17}$ atoms/g H_2O}

σ_{th} = microscopic effective thermal cross section, cm^2
{$\sigma_{th}(O^{17}) = 1.21 \times 10^{-25}$, $\sigma_{th}(N^{14})=9.51 \times 10^{-25}$}

σ_f = microscopic effective fast cross section, cm^2
{$\sigma_f(O^{17}) = 4.79 \times 10^{-26}$, $\sigma_f(N^{14})=3.92\times 10^{-26}$}

ϕ_{th} = thermal neutron flux, 6.32×10^{13} n/cm^2-sec

ϕ_f = fast neutron flux, 3.06×10^{14} n/cm^2-sec

The production rate of the C-14 by $O^{17}(n, \alpha)C^{14}$ reaction is 7.4×10^{11} Bq/cycle. The production rate of the C-14 by $N^{14}(n, p)C^{14}$ reaction is 3.0×10^{10} Bq/cycle. The production rate of C-14 from these sources during reactor operation is 7.7×10^{11} Bq/cycle.

11.1.3.3 Argon-41 Production and Releases

Argon-41 (Ar-41) is formed in the reactor containment building air by neutron activation of naturally occurring Ar-40 in the air surrounding the reactor vessel and could be produced within the reactor coolant by the Ar-40 dissolved in the primary coolant. Ar-41 is released to the environment via the reactor containment building vent when the reactor containment building is vented or purged. The annual release amount of Ar-41 from a pressurized water reactor is assumed to be 34 Ci/yr (Reference 7).

11.1.3.4 Nitrogen-16 Production

Nitrogen-16 (N-16) is produced by the neutron reaction with oxygen-16. N-16 is not a significant radiation source outside the reactor containment building due to its short half-life (7.13 seconds). N-16 activities for the shielding design inside the reactor containment building are provided in Subsection 12.2.1.1.2.

11.1.4 Tritium Production in Reactor Coolant

The principal sources of tritium production in a pressurized water reactor (PWR) are from ternary fission and neutron-induced reactions in boron, lithium, and deuterium that are present in the reactor coolant and control element assemblies (CEAs). The tritium produced in the reactor coolant contributes immediately to the overall tritium concentration, while the tritium produced by fission and neutron capture in the CEAs contributes to the overall tritium concentration via release through the fuel cladding.

11.1.4.1 Activation Sources of Tritium

The activation reactions producing tritium are shown in Table 11.1-15. The tritium production from B-11 and N-14 sources is insignificant due to low reaction cross section and abundance and can be neglected. The activation reactions from B-10, lithium, and deuterium are the major sources of tritium in the reactor coolant and CEAs.

The tritium production from the above sources is determined by the following expressions:

$$\frac{dN}{dt} = \Sigma_a \phi - \lambda N \qquad \text{(Eq. 11.1-13)}$$

$$N = \frac{\Sigma_a \phi}{\lambda}\left(1 - e^{-\lambda t}\right)$$

$$\text{Activity (Bq)} = V\lambda N = \Sigma_a \phi \left(1 - e^{-\lambda t}\right) V$$

Where:

N	=	tritium concentration, atoms/cm^3
$\Sigma_a \phi$	=	production rate, atoms/cm^3-sec
λ	=	decay constant, sec^{-1}
t	=	reactor operating period of interest, sec
V	=	effective core volume or CEA volume, cm^3

The parameters used in the calculation are shown in Table 11.1-16. Based on these parameters, the tritium produced from activation sources in the reactor coolant is provided in Table 11.1-17.

11.1.4.2 Tritium from Fission

The ternary fission production of tritium in the core is calculated using the ORIGEN-S Computer Code (Reference 8). Tritium as a product of fission is released to the reactor coolant through the fuel cladding. One percent of an average expected tritium release from the fuel and 2 percent of a maximum design value are used to estimate the tritium production in the reactor coolant. Tritium production is shown in Table 11.1-17.

11.1.4.3 Tritium Concentrations in the Secondary System

In determining the tritium activity concentrations in the secondary system, it is assumed that tritium that enters the secondary system from the primary system via steam generator tube leakage is uniformly mixed in the secondary system steam and liquid masses. In the equilibrium condition, the decay and leakage losses of tritium from the secondary system are equal to the primary-to-secondary system tritium leakage.

The tritium activity concentrations in the secondary system are calculated by the following equation:

$$N_s = \frac{R}{L_s} N_w \qquad \text{(Eq. 11.1-14)}$$

Where:

N_w = tritium activity concentrations in the primary system, Bq/g

R = primary-to-secondary leak rate, g/sec

L_s = steam leak rate, g/sec

N_s = tritium activity concentrations in the secondary system, Bq/g

11.1.5 Leakage Sources

Systems containing radioactive liquids and gases are potential sources of leakage and discharge to the environment. Liquid leakage is from potential sources such as pump seals and valve packings. Expected leakage of primary coolant into the reactor containment building is at a rate that would result in the release of 3 percent per day of the primary coolant noble gas inventory and 8.0×10^{-4} percent per day of the primary coolant iodine inventory. The expected primary coolant leak rate into the auxiliary building is 72.6 kg/day (160 lb/day), and the expected leak rate of steam into the turbine generator building is 771 kg/hr (1,700 lb/hr). The expected primary-to-secondary leakage rate across the steam generator tubes is 34 kg/day (75 lb/day). Table 11.1-18 provides maximum anticipated leak rates from NSSS-related valves and pumps.

Liquid radioactive releases are further addressed in Section 11.2. Gaseous radioactive effluents to the environment are further addressed in Section 11.3. Concentrations of airborne radioactive nuclides in cubicles are addressed in Subsection 12.2.2.3.

11.1.6 Combined License Information

No combined license (COL) information is required with regard to Section 11.1.

11.1.7 References

1. ANSI/ANS 18.1, “Radioactive Source Term for Normal Operation of Light Water Reactors,” American Nuclear Society, 1999 (withdrawn 2009).

2. 10 CFR 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.”

3. NRC RG 1.112, Revision 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” U.S. Nuclear Regulatory Commission, 2007.

4. P. D. Maloney. “DAMSAM: A Digital Computer Program to Calculate Primary and Secondary Activity Transients,” Combustion Engineering, Inc., 1972.

5. J. D. Eichenberg et al., WAPD-183, "Effects of Irradiation on Bulk UO_2," Bettis Plant, October 1957.

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Table 11.1-1

Parameter Values Used to Calculate
the Reactor Coolant Fission Product Source Term

Parameter	Maximum[(1)]	Normal[(2)]
Core power level (MWt)	4,063	3,983
Duration of reactor operation (core cycles)	5	—
Equilibrium fuel cycle (effective full-power days)	480	—
Thermal neutron flux, n/cm^2-sec	6.32E+13	—
Average thermal fission rate (fission/MW-sec)	3.12E+16	—
Fraction of fuel defect (fraction)	0.01	—
Reactor coolant mass, kg (lb)	2.92E+05 (6.43E+05)	2.92E+05 (6.43E+05)
Core-to-reactor coolant volume ratio (fraction)	0.073	—
Purification flow, kg/sec (lb/sec)	5.02 (11.07)	5.02 (11.07)
Purification flow for boron control, kg/sec (lb/sec), cycle average	—	2.65E-02 (5.85E-02)
Boron concentration at BOC (ppm), minimum	1,110	—
Ion exchanger and gas stripper removal efficiency		
CVCS purification ion exchanger		
Xe, Kr, tritium	0.0	0.0
Cs, Rb	0.5	0.5
Anion	0.99	0.99
Others	0.98	0.98
CVCS gas stripper		
Xe, Kr	0.999	—
Others	0.0	—
CVCS gas stripper operation	Continuous	None
Fission product escape rate coefficients (sec^{-1})		
Xe, Kr	6.5E-08	—
I, Br, Rb, Cs	1.3E-08	—
Mo	2.0E-09	—
Te	1.0E-09	—
Sr, Ba	1.0E-11	—
Y, Zr, Nb, Tc, Ru, La, Ce	1.6E-12	—

(1) Design Basis Source Term (1 % Fuel Defect, DAMSAM Code Input)

(2) Expected Source Term (ANSI/ANS 18.1)

Table 11.1-2

Maximum Reactor Coolant Fission Product Source Term
(Core Power: 4,063 MWt, 1.0 % Fuel Defect, Continuous Gas Stripping)

Nuclide	Specific Activity (Bq/g)	Nuclide	Specific Activity (Bq/g)	Nuclide	Specific Activity (Bq/g)
Kr-85m	3.00E+04	Cs-136	1.89E+03	Nb-95	2.04E+01
Kr-85	7.40E+02	Cs-137	1.63E+04	Mo-99	1.11E+04
Kr-87	2.92E+04	N-16	8.22E+06(3)	Tc-99m	6.66E+03
Kr-88	7.40E+04	H-3	1.30E+05(4)	Ru-103	7.03E+00
Xe-131m	7.40E+03	Na-24	1.81E+03(1)	Ru-106	3.00E+00
Xe-133m	1.92E+03	Cr-51	5.48E+02	Ag-110m	5.15E+01(1)
Xe-133	9.62E+05	Mn-54	6.34E+01(1)	Te-129m	2.37E+02
Xe-135m	2.29E+04	Fe-55	4.75E+01(1)	Te-129	2.52E+02
Xe-135	1.30E+05	Fe-59	1.19E+01(1)	Te-131m	1.11E+03
Xe-137	5.55E+03	Co-58	1.82E+02(1)	Te-131	4.44E+02
Xe-138	1.96E+04	Co-60	2.10E+01(1)	Te-132	7.77E+03
Br-84	7.77E+02	Zn-65	2.02E+01(1)	Ba-137m	1.55E+04
I-131	9.99E+04	Sr-89	1.30E+02	Ba-140	1.59E+02
I-132	2.66E+04	Sr-90	8.88E+00	La-140	5.55E+01
I-133	1.41E+05	Sr-91	1.92E+02	Ce-141	5.92E+00
I-134	1.67E+04	Y-91m	1.11E+02	Ce-143	1.67E+01
I-135	7.77E+04	Y-91	1.89E+01	Ce-144	1.70E+01
Rb-88	7.40E+04	Y-93	4.44E+00	W-187	9.70E+01(1)
Cs-134	1.41E+04	Zr-95	2.40E+01(2)	Np-239	8.62E+01(1)

(1) Based on expected source term
(2) Summation of fission and corrosion product specific activity
(3) Specific activity at the reactor vessel outlet nozzle
(4) Based on the tritium measurement in domestic operating reactors in Korea

Table 11.1-3

Assumptions Used in Determining Activities
in the Spent Fuel Pool Cooling and Cleanup System

1. During the 2 days after shutdown, the primary coolant is purified by the purification filter, purification ion exchanger, and gas stripper of the CVCS. The purified primary coolant is then diluted by the spent fuel and refueling pool cooling water.

2. During the first 30 days after shutdown, the primary coolant, spent fuel pool cooling water, and refueling pool cooling water are simultaneously purified by the spent fuel pool cooling and cleanup system; thereafter, the spent fuel pool cooling water is only purified.

3. The capacity of the spent fuel pool cooling and cleanup system is 1,324.9 L/min (350 gpm).

4. The decontamination factors of the spent fuel pool cooling and cleanup system demineralizers are as follows:

Noble gases	1
I, Br	100
Cs, Rb	2
Others	100

5. No credit is taken for removal of activity by filters in the spent fuel pool cooling and cleanup system when calculating the spent fuel pool and refueling pool cooling water activities. In calculating the cumulative activities of filters, the filter element is assumed to have a decontamination factor of 10.

Table 11.1-4

Maximum and Expected Specific Activities
in the Spent Fuel Pool and Refueling Pool (Bq/g)

Nuclide	Maximum[(1)]	Expected[(2)]`
H-3	4.0E+04	1.1E+04
N-16	0.0E+00	0.0E+00
Kr-85m	0.0E+00	0.0E+00
Kr-85	0.0E+00	0.0E+00
Kr-87	0.0E+00	0.0E+00
Kr-88	0.0E+00	0.0E+00
Xe-131m	0.0E+00	0.0E+00
Xe-133m	0.0E+00	0.0E+00
Xe-133	0.0E+00	0.0E+00
Xe-135m	0.0E+00	0.0E+00
Xe-135	0.0E+00	0.0E+00
Xe-137	0.0E+00	0.0E+00
Xe-138	0.0E+00	0.0E+00
Br-84	0.0E+00	0.0E+00
Rb-88	0.0E+00	0.0E+00
Sr-89	1.1E+00	4.4E–02
Sr-90	3.5E–01	1.9E–02
Sr-91	6.4E–02	1.2E–02
Y-91m	0.0E+00	0.0E+00
Y-91	2.5E+00	2.6E–02
Y-93	6.6E–03	2.4E–01
Zr-95	2.1E–01	1.3E–01
Nb-95	1.5E–01	8.3E–02
Tc-99m	3.3E–01	8.9E–03
Mo-99	5.3E+01	1.2E+00
Ru-103	5.4E–02	2.3E+00
Ru-106	6.5E–02	7.6E+01
Ag-110m	9.1E–01	8.9E–01
Te-129m	1.8E+00	5.5E–02

Nuclide	Maximum[(1)]	Expected[(2)]
Te-129	0.0E+00	0.0E+00
I-131	6.3E+02	5.2E–01
Te-131m	2.9E+00	1.5E–01
Te-131	0.0E+00	0.0E+00
Te-132	3.9E+01	3.4E–01
I-132	3.8E–04	3.2E–05
I-133	2.5E+02	1.8E+00
I-134	0.0E+00	0.0E+00
Cs-134	3.2E+02	3.6E–02
I-135	6.5E+00	1.8E–01
Cs-136	2.4E+01	4.7E–01
Cs-137	4.4E+02	6.2E–02
Ba-140	1.1E+00	3.5E+00
La-140	2.0E–01	3.5E+00
Ce-141	4.4E–02	4.4E–02
Ce-143	5.0E–02	3.3E–01
Ce-144	3.2E–01	2.9E+00
Na-24	1.8E+00	1.8E+00
Cr-51	4.0E+00	9.0E–01
Mn-54	1.3E+00	1.2E+00
Fe-55	1.5E+00	1.5E+00
Fe-59	9.4E–02	9.3E–02
Co-58	1.6E+00	1.6E+00
Co-60	7.4E–01	7.3E–01
Zn-65	3.5E–01	3.4E–01
Ba-137m	4.4E+02	6.2E–02
W-187	2.1E–01	2.1E–01
Np-239	3.8E–01	3.8E–01

(1) Design Basis Source Term (1 % Fuel Defect)

(2) Expected Source Term (ANSI/ANS 18.1)

Table 11.1-5

Assumptions Used in Determining Secondary System Activities

1. Primary coolant activities are described in Subsection 11.1.1.1 for the design basis case and in Subsection 11.1.2.1 for the expected case.

2. Primary-to-secondary leak rates:

Design basis	3,270 L/day (0.6 gal/min)
Expected	34 kg/day (75 lb/day)

3. Flow rates in the secondary system (based on the two steam generators):

Steam flow rate, kg/hr (lb/hr)	8.14×10^6 (1.80×10^7)
Continuous blowdown rate, kg/hr (lb/hr)	1.63×10^5 (3.59×10^5)
High-capacity blowdown rate (hot leg), kg/sec (lb/sec)	8.18×10^1 (1.80×10^2)

4. Liquid masses in the secondary system of two steam generators, kg (lb): 2.41×10^5 (5.32×10^5)

5. Steam generator internal partition coefficients (Reference 7):

H-3	1.0
I, B	0.01
Others	0.005

All noble gases are assumed to be in the steam.

6. Fractions of radionuclide in the main steam reaching the main condenser (Reference 7):

I, Br	0.2
Noble gases	1.0
Others	0.1

7. Decontamination factors of the blowdown demineralizer and condensate polishing demineralizer (Reference 7):

Demineralizer	Decontamination Factor				
	Noble Gases	I, Br	Cs, Rb	H-3	Others
Blowdown	1	100	100	1	100
Condensate polishing	1	10	2	1	10

Table 11.1-6

Design Basis Radionuclide Concentrations
in the Secondary System (Bq/g) (1 % Fuel Defect)

Nuclide	Steam Generator	
	Liquid	Steam
Kr-85m	—	3.71E-01
Kr-85	—	9.16E-03
Kr-87	—	3.61E-01
Kr-88	—	9.16E-01
Xe-131m	—	9.16E-02
Xe-133m	—	2.38E-02
Xe-133	—	1.19E+01
Xe-135m	—	2.83E-01
Xe-135	—	1.61E+00
Xe-137	—	6.87E-02
Xe-138	—	2.43E-01
Br-84	3.03E-01	3.03E-03
I-131	1.20E+02	1.20E+00
I-132	2.17E+01	2.17E-01
I-133	1.62E+02	1.62E+00
I-134	8.86E+00	8.86E-02
I-135	8.08E+01	8.08E-01
Rb-88	1.93E+01	9.64E-02
Cs-134	1.91E+01	9.56E-02
Cs-136	2.55E+00	1.28E-02
Cs-137	2.21E+01	1.11E-01
Cr-51	6.78E-01	3.39E-03
Mn-54	7.85E-02	3.93E-04
Fe-55	5.88E-02	2.94E-04
Fe-59	1.47E-02	7.36E-05
Co-58	2.25E-01	1.13E-03
Co-60	2.60E-02	1.30E-04
Zr-95	2.97E-02	1.49E-04
Zn-65	2.50E-02	1.25E-04

Nuclide	Steam Generator	
	Liquid	Steam
N-16	7.38E-01	3.69E-03
Na-24	2.08E+00	1.04E-02
Sr-89	1.61E-01	8.04E-04
Sr-90	1.10E-02	5.50E-05
SR-91	2.13E-01	1.06E-03
Y-91m	5.81E-02	2.91E-04
Y-91	2.34E-02	1.17E-04
Y-93	4.95E-03	2.47E-05
Nb-95	2.52E-02	1.26E-04
Mo-99	1.35E+01	6.76E-02
Tc-99M	6.93E+00	3.46E-02
Ru-103	8.70E-03	4.35E-05
Ru-106	3.72E-03	1.86E-05
Ag-110m	6.38E-02	3.19E-04
Te-129m	2.93E-01	1.47E-03
Te-129	1.57E-01	7.84E-04
Te-131m	1.32E+00	6.62E-03
Te-131	1.48E-01	7.38E-04
Te-132	9.49E+00	4.74E-02
Ba-137m	6.92E-01	3.46E-03
Ba-140	1.96E-01	9.81E-04
La-140	6.69E-02	3.34E-04
Ce-141	7.32E-03	3.66E-05
Ce-143	2.00E-02	1.00E-04
Ce-144	2.11E-02	1.05E-04
W-187	1.15E-01	5.73E-04
Np-239	1.05E-01	5.23E-04
H-3	1.69E+04	1.69E+04

Table 11.1-7

Radionuclide Crud Concentrations
in the High-Capacity Blowdown Liquid (Bq/g)

Nuclide	Design Basis Source	Expected Source
Cr-51	3.26E+02	3.12E+01
Mn-54	2.43E+04	2.75E+00
Fe-55	2.73E+03	5.35E+01
Fe-59	2.82E+04	3.93E+00
Co-58	8.71E+02	1.95E+02
Co-60	1.09E+02	1.09E+02
Zr-95	8.21E+01	8.21E+01
Zn-65	1.95E+01	1.95E+01

Table 11.1-8

Design Basis Radionuclide Concentrations of Sources to GWMS (Bq/cm^3)(1)

Nuclide	Reactor Drain Tank(2)	Volume Control Tank	Gas Stripper(2)	Equipment Drain Tank(3)
H-3	1.0E+01	1.0E+01	1.7E+01	1.0E+00
Br-84	7.6E–01	1.0E–02	2.6E–01	7.2E–03
Kr-85m	2.7E+04	1.0E+03	1.0E+06	2.5E+03
Kr-85	6.6E+02	2.7E+01	2.5E+04	6.2E+01
Kr-87	2.6E+04	8.9E+02	9.8E+05	2.5E+03
Kr-88	6.6E+04	2.5E+03	2.5E+06	6.2E+03
Xe-131m	6.6E+03	1.5E+02	2.5E+05	6.2E+02
Xe-133m	1.7E+03	3.9E+01	6.4E+04	1.6E+02
Xe-133	8.6E+05	2.0E+04	3.2E+07	8.1E+04
Xe-135m	2.0E+04	2.5E+02	7.7E+05	1.9E+03
Xe-135	1.2E+05	2.5E+03	4.4E+06	1.1E+04
Xe-137	4.9E+03	2.7E+01	1.9E+05	4.7E+02
Xe-138	1.7E+04	2.0E+02	6.6E+05	1.7E+03
I-131	9.8E+01	1.7E+00	3.3E+01	9.3E–01
I-132	2.6E+01	4.2E–01	8.9E+00	2.5E–01
I-133	1.4E+02	2.3E+00	4.7E+01	1.3E+00
I-134	1.6E+01	2.4E–01	5.6E+00	1.5E–01
I-135	7.6E+01	1.2E+00	2.6E+01	7.2E–01

(1) 1.0 % fuel defect and continuous gas stripping are applied.

(2) Reactor drain tank and gas stripper specific activities are based on continuous venting at 0.680 L/m (0.024 scfm) and 9.061 L/m (0.32 scfm) to the GWMS.

(3) Equipment drain tank specific activities are based on continuous venting at 0.14 L/m (0.005 scfm) to the GWMS.

Table 11.1-9

Expected Specific Activities of Reactor Coolant During Normal Operation[1]
(Core Power: 3,983 MWt, No Gas Stripping)

Nuclide	Specific Activity (Bq/g)	Nuclide	Specific Activity (Bq/g)	Nuclide	Specific Activity (Bq/g)
Kr-85m	5.96E+02	Cs-136	3.70E+01	Nb-95	1.11E+01
Kr-85	4.33E+04	Cs-137	2.27E+00	Mo-99	2.51E+02
Kr-87	6.32E+02	N-16	1.48E+06[2]	Tc-99m	1.79E+02
Kr-88	6.70E+02	H-3	3.70E+04[3]	Ru-103	2.97E+02
Xe-131m	3.27E+04	Na-24	1.81E+03	Ru-106	3.57E+03
Xe-133m	2.71E+03	Cr-51	1.23E+02	Ag-110m	5.15E+01
Xe-133	1.18E+03	Mn-54	6.34E+01	Te-129m	7.52E+00
Xe-135m	4.83E+03	Fe-55	4.75E+01	Te-129	8.97E+02
Xe-135	2.51E+03	Fe-59	1.19E+01	Te-131m	5.84E+01
Xe-137	1.26E+03	Co-58	1.82E+02	Te-131	2.87E+02
Xe-138	2.27E+03	Co-60	2.10E+01	Te-132	6.68E+01
Br-84	5.97E+02	Zn-65	2.02E+01	Ba-137m	2.27E+00
I-131	8.23E+01	Sr-89	5.54E+00	Ba-140	5.14E+02
I-132	2.27E+03	Sr-90	4.75E-01	La-140	9.77E+02
I-133	1.04E+03	Sr-91	3.67E+01	Ce-141	5.94E+00
I-134	3.74E+03	Y-91m	1.72E+01	Ce-143	1.09E+02
I-135	2.13E+03	Y-91	2.06E-01	Ce-144	1.58E+02
Rb-88	7.07E+03	Y-93	1.61E+02	W-187	9.70E+01
Cs-134	1.59E+00	Zr-95	1.54E+01	Np-239	8.62E+01

(1) Expected Source Term (ANSI/ANS 18.1)

(2) Specific activity at the reactor coolant entering the letdown line

(3) The concentration of tritium is a function of the inventory of tritiated liquids in the plant, rate of production of tritium due to activation in the reactor coolant, rate of release from the fuel, and extent to which tritiated water is recycled or discharged from the plant. The value of tritium concentration listed in this table is typical in PWRs with the assumption that a moderate amount of tritium is recycled (Reference 1).

Table 11.1-10

Expected Radionuclide Concentrations in the Secondary System (Bq/g)

Nuclide	Steam Generator	
	Liquid	Steam
Kr-85M	—	1.04E-04
Kr-85	—	7.54E-03
Kr-87	—	1.10E-04
Kr-88	—	1.17E-04
Xe-131m	—	5.69E-03
Xe-133 m	—	4.72E-04
Xe-133	—	2.05E-04
Xe-135 m	—	8.41E-04
Xe-135	—	4.37E-04
Xe-137	—	2.19E-04
Xe-138	—	3.95E-04
Br-84	3.28E-03	3.28E-05
I-131	1.39E-03	1.39E-05
I-132	2.61E-02	2.61E-04
I-133	1.68E-02	1.68E-04
I-134	2.79E-02	2.79E-04
I-135	3.12E-02	3.12E-04
Rb-88	2.59E-02	1.30E-04
Cs-134	3.03E-05	1.52E-07
Cs-136	7.03E-04	3.52E-06
Cs-137	4.33E-05	2.17E-07
Cr-51	2.14E-03	1.07E-05
Mn-54	1.10E-03	5.52E-06
Fe-55	8.28E-04	4.14E-06
Fe-59	2.07E-04	1.04E-06
Co-58	3.17E-03	1.58E-05
Co-60	3.66E-04	1.83E-06
Zr-95	2.68E-04	1.34E-06
Zn-65	3.52E-04	1.76E-06

Nuclide	Steam Generator	
	Liquid	Steam
N-16	1.87E-03	9.35E-06
Na-24	2.93E-02	1.47E-04
Sr-89	9.65E-05	4.82E-07
Sr-90	8.28E-06	4.14E-08
Sr-91	5.72E-04	2.86E-06
Y-91m	1.27E-04	6.33E-07
Y-91	3.59E-06	1.79E-08
Y-93	2.52E-03	1.26E-05
Nb-95	1.93E-04	9.66E-07
Mo-99	4.30E-03	2.15E-05
Tc-99m	2.62E-03	1.31E-05
Ru-103	5.17E-03	2.58E-05
Ru-106	6.22E-02	3.11E-04
Ag-110m	8.97E-04	4.49E-06
Te-129m	1.31E-04	6.54E-07
Te-129	7.86E-03	3.93E-05
Te-131m	9.81E-04	4.90E-06
Te-131	1.34E-03	6.71E-06
Te-132	1.15E-03	5.74E-06
Ba-137m	1.43E-06	7.13E-09
Ba-140	8.92E-03	4.46E-05
La-140	1.66E-02	8.28E-05
Ce-141	1.03E-04	5.17E-07
Ce-143	1.84E-03	9.18E-06
Ce-144	2.75E-03	1.38E-05
W-187	1.61E-03	8.07E-06
Np-239	1.47E-03	7.36E-06
H-3	6.81E+01	6.81E+01

Table 11.1-11

Long-Lived Isotopes in Crud

Isotope	Half Life	λ (d^{-1})	Parent	Reaction	γ/dis(1)	E (MeV)
Cr-51	27.70 days	2.50E-02	Cr-50	n, γ	0.1	0.32
Mn-54	312.3 days	2.22E-03	Fe-54	n, p	1	0.84
Fe-59	44.50 days	1.56E-02	Fe-58	n, γ	1	1.18
Co-60	5.272 years	3.60E-04	Co-59	n, γ	2	1.25
Co-58	70.82 days	9.77E-03	Ni-58	n, p	1	0.81
Zr-95	64.02 days	1.08E-02	Zr-94	n, γ	2	0.75

(1) gamma/disintegration

Table 11.1-12

Parameters for Crud Activity

Parameter	Value
Thermal neutron flux, n/cm^2-sec	6.32E+13
Fast neutron flux, n/cm^2-sec	3.06E+14
RCS surface area / core surface area, A_T/A_C	4.8

Core Residence Times and Activation Rates

Isotope	Core Residence Time (t_{res}, day)	Activation Rate (reactions/g-sec)
Cr-51	12	1.34E+11
Mn-54	110	4.37E+08
Fe-59	Saturated	1.99E+08
Co-58	23	4.18E+10
Co-60	197	4.32E+09
Zr-95	29	8.65E+08

Table 11.1-13

Long-Lived Crud Activity

Isotope	Half Life	Activity (Bq/g-crud)
Cr-51	27.70 days	7.31E+09
Mn-54	312.3 days	1.99E+07
Fe-59	44.50 days	4.18E+07
Co-58	70.82 days	1.77E+09
Co-60	5.272 years	6.22E+07
Zr-95	64.02 days	4.90E+07

Table 11.1-14

Calculated Average Crud Activity
in the Reactor Coolant

Isotope	Activity (Bq/g-coolant)
Cr-51	5.48E+02
Mn-54	1.49E+00
Fe-59	3.14E+00
Co-58	1.33E+02
Co-60	4.66E+00
Zr-95	3.67E+00

Table 11.1-15

Tritium Activation Reactions

Reaction	Threshold Energy, MeV	Cross Section, cm^{2}(1)
$B^{10}(n, 2\alpha)T$	1.4	1.26E-26
$Li^{7}(n, n\alpha)T$	3.9	7.79E-27
$Li^{6}(n, \alpha)T$	Thermal	9.44E-22
$H^{2}(n, \gamma)T$	Thermal	5.50E-28
$B^{11}(n, T)Be^{9}$	10.4	7.30E-30
$N^{14}(n, T)C^{12}$	4.3	3.00E-28

(1) Spectrum averaged value for neutrons of energy greater than 0.625 eV

Table 11.1-16

Parameters Used for Calculating Tritium Production

Parameter	Value
Active core water volume, cm^3	3.00E+07
Thermal neutron flux, n/cm^2-sec	6.32E+13
Fast neutron flux, n/cm^2-sec	3.06E+14
Average lithium concentration in the reactor coolant, ppm	
Expected	2.2
Maximum	3.5
Lithium-6 abundance, a/o	0.1
Average boron concentration in the reactor coolant, ppm	
Expected	652
Maximum	755
Power level, MWt	4,063
Tritium release from fuel, %	
Expected	1.0
Maximum	2.0
Tritium release from CEA, %	50.0

Table 11.1-17

Tritium Production in the Reactor Coolant

Source	Average (Bq/cycle)	Maximum (Bq/cycle)
$H^2(n, \gamma)T$	2.57E+11	2.57E+11
$Li^6(n, \alpha)T$	1.09E+13	1.73E+13
$Li^7(n, n\alpha)T$	6.90E+11	1.10E+12
$B^{10}(n, 2\alpha)T$	4.28E+13	4.95E+13
Fission products(1)	1.04E+13	2.07E+13
CEAs	2.30E+12	1.29E+13
Total	6.73E+13	1.02E+14

(1) Tritium Production from Ternary Fission (ORIGEN-S Code)

Table 11.1-18

Maximum Anticipated Leakage Rates from NSSS-Related Components to the Building Environment

Component	Assumed Leakage Rates
Valves	
Disk leakage	4 cm^3/hr/cm of seat diameter
Stem leakage	4 cm^3/hr/cm of stem diameter
Pumps	
Centrifugal (mechanical seals) (except SI and SC pumps)	50 cm^3/hr per seal during normal operating conditions with availability of seal cooling water
	100 cm^3/hr per seal during loss of externally supplied cooling water
Positive displacement	3,785 cm^3/hr (1 gal/hr)
SI and SC pumps	1,000 cm^3/hr per seal (each pump)
Flanges	30 cm^3/hr