11.0 Radioactive Waste Management

11.1 Source Terms

Section 11.1 presents the sources of radioactivity generated in the reactor and the resulting source terms from normal plant operations including anticipated operational occurrences (AOO). This section provides two source terms for the primary and secondary coolant systems: design and realistic. The design basis source term is obtained by applying conservative assumptions and serves as a basis for the design capacities and performance of the waste management systems, as well as for process and effluent radiological monitoring systems. This source term also forms the basis for the shielding requirements and for demonstrating compliance with occupational radiation exposures limits, described further in Chapter 12. The design basis source term also provides the core radionuclide inventory and primary coolant concentrations for the initial conditions in the design basis accidents (DBA) described in Chapter 15. The realistic source term represents the expected average concentrations of radionuclides in the primary and secondary coolant under normal operating conditions. The basis for the realistic source term is industry experience at a large number of operating nuclear power plants.

11.1.1 Sources of Radioactivity: Fission, Activation, and Corrosion Products

Reactor operations generate radioactive fission, activation, and corrosion products. The fission process produces radioactive fission products and neutrons in the core, which activate both the coolant and the corrosion products present in the coolant. Although the radioactive inventory is contained in the fuel rod, during normal operation radioactive products diffuse through the fuel rod cladding or leak through defects in the cladding and enter the reactor coolant system (RCS) coolant. These products can also pass from the RCS to the secondary coolant system through steam generator tube leaks. Fission products typically include noble gases, halogens, rubidium, cesium, and other miscellaneous nuclides.

Within the coolant systems, the activation of oxygen in the water coolant forms activation products. These activation products are independent of fuel defect level; the calculated levels of these products were determined using operating plant data. Specific activation products include the following:

- Nitrogen-16 production results from neutron reaction with oxygen-16. Nitrogen-16 is a significant gamma emitter, but has a short half-life. This chapter presents data for nitrogen-16 but, because of its short half-life, this product is primarily a concern within the containment area. Chapter 12 addresses nitrogen-16 in detail; specifically, Table 12.2-4 provides nitrogen-16 concentrations at various locations in the RCS.

- Nitrogen-17 production results from the neutron reaction with oxygen-17. Nitrogen-17 is a short-lived neutron emitter and therefore is not a significant radiation source.

- Tritium is continuously produced during reactor operation from ternary fission in the fuel and from neutron activation of reactor coolant materials. In the reactor coolant, approximately two-thirds of the tritium activity results from the activation of boron-10 in the soluble poison, and approximately one-third results from the leakage and diffusion of ternary fission tritium through the fuel clad. Less than three percent of tritium activity is produced from the activation of lithium and deuterium in the reactor coolant.

- Argon-41 production occurs by neutron activation of naturally occurring argon in air. Argon-41 production is primarily within the biological shield.

Carbon-14 is formed by the activation of carbon, nitrogen or oxygen. The primary reactions in a light water reactor are: $^{13}C(n,\gamma)\ ^{14}C$, $^{14}N(n,p)\ ^{14}C$, and $^{17}O(n,\alpha)\ ^{14}C$.

Reactor coolant corrosion products result from the in-core activation of impurities in the coolant water and activation of corrosion and wear products (including metallic fines) circulating in the RCS. Corrosion products include isotopes of sodium, iron, cobalt and other miscellaneous nuclides. Corrosion product calculations are also based on operating plant data and are independent of fuel defect level.

11.1.2 Design Basis

The source terms described in this section are used to demonstrate that U.S. EPR plant effluent concentrations are below 10 CFR Part 20, Appendix B, Table 2 concentrations and that the radwaste system design meets the as low as reasonably achievable (ALARA) design objectives of 10 CFR Part 50 Appendix I. These source terms are determined following the guidance in Section 11 of NUREG-0800 (Reference 1). The source terms and the evaluations presented here demonstrate that the plant design meets GDC 60, which requires that the nuclear power unit design include a means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including AOOs. GDC 60 also requires that sufficient holdup capacity be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

11.1.2.1 Design Basis for Radwaste System and Normal Effluents

Two sets of source terms that describe reactor coolant radionuclide concentrations have been determined for the evaluation of radwaste systems and effluent releases. The first is a conservative design basis source term based on the assumption that the primary coolant radionuclide concentrations are made up of a combination of

Technical Specification (TS) limits for halogens (1 μCi/gm dose equivalent (DE) iodine-131 in primary coolant and 0.1 μCi/gm in the secondary coolant) and noble gases (210 μCi/gm DE xenon-133) in primary coolant. Activation product and tritium concentrations are derived from the ANSI/ANS 18.1-1999 standard (Reference 2). Since the activated corrosion products are independent of failed fuel fraction, design basis and realistic basis concentrations for corrosion products are assumed to be the same. Design basis values for the remaining fission product radionuclides are calculated based on a 1.0 percent failed fuel fraction. Design basis secondary coolant concentrations are based on the TS limit primary to secondary leak rate, which totals 600 gpd from the four steam generators.

The second source term is developed using a realistic model in which the reactor coolant radionuclide concentrations are based on observed industry experience. The model used is described in RG 1.112, Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Reactors, with the source term calculated using NUREG-0017 (Reference 3), which contains the PWR GALE Code, revised 1985.

11.1.2.2 Design Basis for Shielding

The source term used for the shielding evaluation in Chapter 12 is based on a failed fuel fraction assumption that differs from the source term used for the radwaste effluent evaluation. The shielding evaluation source term for normal operations is based on a radionuclide concentration in liquid and gases that is determined using 0.25 percent failed fuel fraction for the nuclides except for radioiodines and bromines. For dose equivalent (DE) I-131, the TS concentration limit is applied, which is more conservative than the value associated with the 0.25 percent failed fuel fraction. An adjustment factor, based on the increased DE I-131 concentration, is also applied to I-129 and I-130, which are not included in the DE I-131 definition, and to the bromines. The 0.25 percent failed fuel fraction is consistent with SRP Section 12.2 (Reference 1). The source term used for post-accident shielding evaluations for vital area access is based on RG 1.183.

The shielding source term is also used for the initial RCS radionuclide concentration for the DBA evaluations in Chapter 15. The DBA source terms are determined in Chapter 15 following the guidance in the SRP Section 15.0.3 (Reference 1) and RG 1.183.

11.1.2.3 Reactor Coolant System Design Basis Source Term

The design basis RCS activity was developed from a bounding core radionuclide inventory using the ORIGEN-2.1 software (Reference 4) along with extended burnup libraries from ORIGEN-2 burnup reactor models (Reference 5). U.S. EPR-specific parameters used in the calculation are listed in Table 11.1-1—Parameters Used to

Calculate RCS Design Source Term Activity. Bounding values of core inventory radionuclides are listed in Table 11.1-2—RCS Design Basis Source Term.

A maximum core average radionuclide inventory was calculated from a parametric evaluation with a fuel enrichment of 5 wt% uranium-235 and burnup steps ranging between 5 and 41 GWD/MTU. The maximum activity for each radionuclide from the parametric cases was selected to provide a maximum core average inventory. The results of ORIGEN-2 calculations and the U.S. EPR-specific parameters listed in Table 11.1-1 were then used to determine a design basis reactor coolant activity for this chapter. Table 11.1-2 lists the resulting RCS radionuclide concentrations for both the shielding and effluent source terms.

11.1.2.4 Secondary Coolant Design Basis Source Term

The design basis activity input into the secondary side is 600 gpd of reactor coolant, with design basis radionuclides as determined above. U.S. EPR-specific design inputs are listed in Table 11.1-3—Parameters Used to Calculate Secondary Coolant Design Basis Source Terms. Design basis secondary coolant activity concentrations are shown in Table 11.1-4—Secondary Coolant Design Basis Source Term, Liquid Concentrations and Table 11.1-5—Secondary Coolant Design Basis Source Term, Steam Concentrations for shielding and effluent source terms.

11.1.3 Reactor Coolant System and Secondary Coolant Realistic Source Terms

Realistic source terms for reactor coolant and secondary coolant activity were calculated using the model described in ANSI/ANS 18.1-1999 (Reference 2). Radionuclide coolant concentrations specified in Reference 2 for a reference pressurized water reactor with U-tube steam generators were adjusted in accordance with this standard to reflect the operating-parameter differences between the U.S. EPR and the reference plant. Table 11.1-6—Parameters Used to Calculate Realistic Source Terms provides a comparison of the U.S. EPR parameters with those in the Reference 2 standard. Realistic RCS and secondary source terms are listed in Table 11.1-7—RCS and Secondary Coolant System Realistic Source Terms. RCS and secondary system source terms represent radioactive liquid and gaseous material that may be transported or released to the environment by radioactive waste systems. These liquid and gaseous effluents are determined using the model described in NUREG-0017 PWR-GALE code (Reference 3). Details of the modeling, such as leakage mechanisms, leakage flowrates, release pathways, and outputs obtained are presented in Sections 11.2 and 11.3.

11.1.4 References

1. NUREG-0800, "U.S. NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 2007.

2. ANS/ANSI-18.1-1999, "American National Standard-Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society/American National Standards Institute, September 21, 1999.

3. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors PWR-GALE Code," Revision 1, U.S. Nuclear Regulatory Commission, April 1985.

4. RSIC Computer Code Collection CCC-371, "ORIGEN 2.1-Isotope Generation and Depletion Code-Matrix Exponential Method," Oak Ridge National Laboratory, August 1991.

5. ORNL/TM-11018, "Standard and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN-2 Computer Code," Oak Ridge National Laboratory, December 1989.

Table 11.1-1—Parameters Used to Calculate RCS Design Source Term Activity
Sheet 1 of 2

<table>
<tr><th>PARAMETER</th><th colspan="2">VALUE</th></tr>
<tr><td>Total core thermal power (MW_t) for design-basis applications, including measurement uncertainty</td><td colspan="2">4590 + 22 = 4612 MW_t</td></tr>
<tr><td>Number of fuel assemblies in core</td><td colspan="2">241</td></tr>
<tr><td>Fuel enrichment U-235 $^w/_0$</td><td colspan="2">2%–5%</td></tr>
<tr><td>Mass of UO_2 in fuel assembly</td><td colspan="2">607 kg</td></tr>
<tr><td>Maximum fuel assembly burn-up</td><td colspan="2">62 GWD/MTU</td></tr>
<tr><td rowspan="7">Escape rate coefficients</td><td>Element</td><td>Escape Rate Coefficient (s^{-1})</td></tr>
<tr><td>Xe, Kr, gases</td><td>6.5E-08</td></tr>
<tr><td>I, Br, Rb, Cs</td><td>1.3E-08</td></tr>
<tr><td>Mo</td><td>2.0E-09</td></tr>
<tr><td>Te</td><td>1.0E-09</td></tr>
<tr><td>Sr, Ba</td><td>1.0E-11</td></tr>
<tr><td>Others</td><td>1.6E-12</td></tr>
<tr><td>Volume of RCS
(not including pressurizer)</td><td colspan="2">13,596 ft^3</td></tr>
<tr><td>RCS average operating temperature</td><td colspan="2">594°F</td></tr>
<tr><td>Primary system operating pressure</td><td colspan="2">2250 psia</td></tr>
<tr><td>RCS water specific volume (at 594°F and 2250 psia)</td><td colspan="2">0.02290 ft^3/lb</td></tr>
<tr><td>Density of reactor coolant water (at 594°F and 2250 psia)</td><td colspan="2">43.65 lb/ft^3 (0.699 g/cm^3)</td></tr>
<tr><td>RCS mass exclusive of pressurizer, at 100% power</td><td colspan="2">594,400 lb</td></tr>
<tr><td>Pressurizer volume</td><td colspan="2">1413 ft^3 (40 m^3)</td></tr>
<tr><td>Density of water in pressurizer</td><td colspan="2">37.07 lb/ft^3</td></tr>
<tr><td>Mass of water in pressurizer</td><td colspan="2">52,380 lb</td></tr>
<tr><td>Total volume of water in RCS and pressurizer</td><td colspan="2">15,009 ft^3</td></tr>
<tr><td>Total mass of water in RCS and Pressurizer</td><td colspan="2">6.47E+05 lb_m</td></tr>
<tr><td>RCS letdown flow rate for purification</td><td colspan="2">36,000 kg/hr (7.94E+04 lb_m/hr)</td></tr>
<tr><td>RCS letdown flow rate for boron control (yearly average)</td><td colspan="2">500 lb_m/hr</td></tr>
<tr><td>Secondary water mass</td><td colspan="2">6.792E+05 lb_m total
(1.698E+05 lb_m/SG)</td></tr>
<tr><td>Secondary water volume</td><td colspan="2">1.476E+04 ft^3 total
(3689.2 ft^3/SG)</td></tr>
</table>

Table 11.1-1—Parameters Used to Calculate RCS Design Source Term Activity
Sheet 2 of 2

PARAMETER	VALUE
Secondary water density	46.0 lb/ft^3 (0.737 g/cm^3)
Secondary steam volume	1.882E+04 ft^3 (4704.4 ft^3/SG)
Secondary steam density	2.56 lb/ft^3 (0.041 g/cm^3)
Fraction of noble gas activity in the letdown stream that is not returned to the RCS (not including the boron recovery system)	0%
Fraction of RCS activity removed in passing through the purification mixed-bed demineralizer	Noble Gases, N-16, H-3: 0.0 Anion: 0.99 Cs, Rb: 0.50 Others: 0.98
Coolant transit time through active fuel	0.87 s
Total transit time through the reactor coolant loop	9.93 s

Table 11.1-2—RCS Design Basis Source Term
Sheet 1 of 4

Nuclide	Reactor Coolant Concentration, Shielding (μCi/g)[1]	Reactor Coolant Concentration, Effluent (μCi/g)[2]
Noble Gases		
Kr-83m	1.3E-01	1.3E-01
Kr-85m	5.7E-01	5.7E-01
Kr-85	5.3E+00	5.3E+00
Kr-87	3.3E-01	3.3E-01
Kr-88	1.0E+00	1.0E+00
Kr-89	2.4E-02	2.4E-02
Xe-131m	1.1E+00	1.1E+00
Xe-133m	1.4E+00	1.4E+00
Xe-133	9.5E+01	9.5E+01
Xe-135m	2.0E-01	2.0E-01
Xe-135	3.4E+00	3.4E+00
Xe-137	4.6E-02	4.6E-02
Xe-138	1.6E-01	1.6E-01
DE Xe-133	210	210
Halogens		
Br-83	3.2E-02	3.2E-02
Br-84	1.7E-02	1.7E-02
Br-85	2.0E-03	2.0E-03
I-129	4.6E-08	4.6E-08
I-130	5.0E-02	5.0E-02
I-131	7.4E-01	7.4E-01
I-132	3.7E-01	3.7E-01
I-133	1.3E+00	1.3E+00
I-134	2.4E-01	2.4E-01
I-135	7.9E-01	7.9E-01
D. E. I-131	1.0	1.0
Rubidium, Cesium		
Rb-86m	3.0E-07	1.2E-06
Rb-86	1.9E-03	7.7E-03
Rb-88	1.0E+00	4.1E+00

Table 11.1-2—RCS Design Basis Source Term
Sheet 2 of 4

Nuclide	Reactor Coolant Concentration, Shielding (µCi/g)[1]	Reactor Coolant Concentration, Effluent (µCi/g)[2]
Rb-89	4.7E-02	1.9E-01
Cs-134	1.7E-01	6.8E-01
Cs-136	5.3E-02	2.1E-01
Cs-137	1.1E-01	4.3E-01
Cs-138	2.2E-01	8.8E-01
Miscellaneous Nuclides		
Sr-89	6.4E-04	2.5E-03
Sr-90	3.3E-05	1.3E-04
Sr-91	1.0E-03	4.1E-03
Sr-92	1.7E-04	6.9E-04
Y-90	7.7E-06	3.1E-05
Y-91m	5.2E-04	2.1E-03
Y-91	8.1E-05	3.2E-04
Y-92	1.4E-04	5.6E-04
Y-93	6.5E-05	2.6E-04
Zr-95	9.3E-05	3.7E-04
Zr-97	6.7E-05	2.7E-04
Nb-95	9.4E-05	3.7E-04
Mo-99	1.1E-01	4.3E-01
Tc-99m	4.6E-02	1.9E-01
Ru-103	7.8E-05	3.1E-04
Ru-105	9.5E-05	3.8E-04
Ru-106	2.7E-05	1.1E-04
Rh-103m	6.8E-05	2.7E-04
Rh-105	4.4E-05	1.8E-04
Rh-106	2.7E-05	1.1E-04
Ag-110m	2.0E-07	7.9E-07
Ag-110	1.1E-08	4.4E-08
Sb-125	8.0E-07	3.2E-06
Sb-127	5.0E-06	2.0E-05
Sb-129	6.8E-06	2.7E-05

Table 11.1-2—RCS Design Basis Source Term
Sheet 3 of 4

Nuclide	Reactor Coolant Concentration, Shielding (µCi/g)[1]	Reactor Coolant Concentration, Effluent (µCi/g)[2]
Te-127m	4.4E-04	1.8E-03
Te-127	2.2E-03	8.7E-03
Te-129m	1.5E-03	5.8E-03
Te-129	2.4E-03	9.6E-03
Te-131m	3.7E-03	1.5E-02
Te-131	2.6E-03	1.0E-02
Te-132	4.1E-02	1.6E-01
Te-134	6.7E-03	2.7E-02
Ba-137m	1.0E-01	4.1E-01
Ba-139	2.2E-02	8.6E-02
Ba-140	6.2E-04	2.5E-03
La-140	1.6E-04	6.4E-04
La-141	5.3E-05	2.1E-04
La-142	3.1E-05	1.3E-04
Ce-141	8.9E-05	3.5E-04
Ce-143	7.6E-05	3.0E-04
Ce-144	6.9E-05	2.8E-04
Pr-143	8.8E-05	3.5E-04
Pr-144	6.9E-05	2.8E-04
Nd-147	3.4E-05	1.4E-04
Np-239	8.7E-04	3.5E-03
Pu-238	2.0E-07	7.9E-07
Pu-239	2.0E-08	8.1E-08
Pu-240	2.8E-08	1.1E-07
Pu-241	6.9E-06	2.8E-05
Am-241	7.8E-09	3.1E-08
Cm-242	1.9E-06	7.5E-06
Cm-244	1.0E-07	4.1E-07
Activation Products		
Na-24	3.7E-02	3.7E-02
Cr-51	2.0E-03	2.0E-03

**Table 11.1-2—RCS Design Basis Source Term
Sheet 4 of 4**

Nuclide	Reactor Coolant Concentration, Shielding (µCi/g)[1]	Reactor Coolant Concentration, Effluent (µCi/g)[2]
Mn-54	1.0E-03	1.0E-03
Fe-55	7.6E-04	7.6E-04
Fe-59	1.9E-04	1.9E-04
Co-58	2.9E-03	2.9E-03
Co-60	3.4E-04	3.4E-04
Zn-65	3.2E-04	3.2E-04
W-187	1.8E-03	1.8E-03
Tritium		
H-3	1.0E+00	4.0E+00

Notes:

1. Concentrations for all nuclides except noble gases, halogens, and activation products, based on a conservative fuel defect level of 0.25% failed fuel fraction.

2. Concentrations for all nuclides except noble gases, halogens, and activation products, based on a conservative fuel defect level of 1.0% failed fuel fraction.

Table 11.1-3—Parameters Used to Calculate Secondary Coolant Design Basis Source Terms

PARAMETER	VALUE
Total steam generator secondary side water inventory	6.79E+05 lb_m total (1.698E+05 lb_m/SG)
Secondary side steam generator steam mass	4.808E+04 lb_m total (1.202E+04 lb_m/SG)
Total steam flow rate	2.07E+07 lb_m/hr (5.17E+06 lb_m/hr/SG)
Steam generator blowdown flow rate (4 SGs)	14.4 lb/s/SG (2.08E+05 lb_m/hr total)
Primary to secondary leak rate	600 gpd
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system	0%
Ratio of condensate demineralizer flow rate to the total steam flow	100%
Fraction of activity removed in passing through the steam generator blowdown demineralizers:	Noble Gases, N-16, H-3: 0.0 Anion: 0.90 Cs, Rb: 0.50 Others: 0.90
Ratio of concentration in steam to that in the steam generator	Noble Gases: N/A Br, I: 0.01 Cs, Rb: 0.005 N-16: N/A H-3: 1.0 Others: 0.005

**Table 11.1-4—Secondary Coolant Design Basis Source Term, Liquid Concentrations
Sheet 1 of 3**

Nuclide	Secondary Coolant Liquid Concentration, Shielding (µCi/g)[1]	Secondary Coolant Liquid Concentration, Effluent (µCi/g)
Halogens		
Br-83	1.6E-03	1.6E-03
Br-84	3.1E-04	3.1E-04
Br-85	3.9E-06	3.9E-06
I-129	4.8E-09	4.8E-09
I-130	4.3E-03	4.3E-03
I-131	7.7E-02	7.7E-02
I-132	2.3E-02	2.3E-02
I-133	1.2E-01	1.2E-01
I-134	6.7E-03	6.7E-03
I-135	6.0E-02	6.0E-02
DE I-131	0.10	0.10
Rubidium, Cesium		
Rb-86m	2.2E-12	9.0E-12
Rb-86	3.8E-06	1.5E-05
Rb-88	1.3E-04	5.0E-04
Rb-89	5.0E-06	2.0E-05
Cs-134	3.4E-04	1.4E-03
Cs-136	1.1E-04	4.2E-04
Cs-137	2.2E-04	8.7E-04
Cs-138	4.7E-05	1.9E-04
Miscellaneous Nuclides		
Sr-89	7.2E-07	2.9E-06
Sr-90	3.6E-08	1.4E-07
Sr-91	9.0E-07	3.6E-06
Sr-92	1.0E-07	4.0E-07
Y-90	9.6E-09	3.8E-08
Y-91m	5.4E-07	2.2E-06
Y-91	9.2E-08	3.7E-07
Y-92	1.3E-07	5.3E-07
Y-93	5.8E-08	2.3E-07

Table 11.1-4—Secondary Coolant Design Basis Source Term, Liquid Concentrations
Sheet 2 of 3

Nuclide	Secondary Coolant Liquid Concentration, Shielding (μCi/g)[1]	Secondary Coolant Liquid Concentration, Effluent (μCi/g)
Zr-95	1.0E-07	4.1E-07
Zr-97	6.5E-08	2.6E-07
Nb-95	1.0E-07	4.2E-07
Mo-99	1.2E-04	4.6E-04
Tc-99m	6.6E-05	2.6E-04
Ru-103	8.6E-08	3.4E-07
Ru-105	7.0E-08	2.8E-07
Ru-106	3.0E-08	1.2E-07
Rh-103m	7.7E-08	3.1E-07
Rh-105	5.0E-08	2.0E-07
Rh-106	3.0E-08	1.2E-07
Ag-110m	2.2E-10	8.8E-10
Ag-110	3.0E-12	1.2E-11
Sb-125	8.9E-10	3.5E-09
Sb-127	5.5E-09	2.2E-08
Sb-129	4.8E-09	1.9E-08
Te-127m	4.9E-07	2.0E-06
Te-127	2.0E-06	8.1E-06
Te-129m	1.6E-06	6.4E-06
Te-129	1.6E-06	6.3E-06
Te-131m	3.8E-06	1.5E-05
Te-131	1.1E-06	4.6E-06
Te-132	4.4E-05	1.8E-04
Te-134	1.6E-06	6.4E-06
Ba-137m	2.0E-04	8.1E-04
Ba-139	9.7E-06	3.9E-05
Ba-140	6.8E-07	2.7E-06
La-140	2.1E-07	8.4E-07
La-141	3.7E-08	1.5E-07
La-142	1.4E-08	5.6E-08
Ce-141	9.9E-08	3.9E-07

Table 11.1-4—Secondary Coolant Design Basis Source Term, Liquid Concentrations
Sheet 3 of 3

Nuclide	Secondary Coolant Liquid Concentration, Shielding (µCi/g)[1]	Secondary Coolant Liquid Concentration, Effluent (µCi/g)
Ce-143	7.9E-08	3.1E-07
Ce-144	7.7E-08	3.1E-07
Pr-143	9.8E-08	3.9E-07
Pr-144	7.7E-08	3.1E-07
Nd-147	3.8E-08	1.5E-07
Np-239	9.3E-07	3.7E-06
Pu-238	2.2E-10	8.9E-10
Pu-239	2.3E-11	9.0E-11
Pu-240	3.1E-11	1.2E-10
Pu-241	7.7E-09	3.1E-08
Am-241	8.7E-12	3.5E-11
Cm-242	2.1E-09	8.3E-09
Cm-244	1.1E-10	4.5E-10
Activation Products		
Na-24	3.5E-05	3.5E-05
Cr-51	2.2E-06	2.2E-06
Mn-54	1.1E-06	1.1E-06
Fe-55	8.5E-07	8.5E-07
Fe-59	2.1E-07	2.1E-07
Co-58	3.2E-06	3.2E-06
Co-60	3.8E-07	3.8E-07
Zn-65	3.6E-07	3.6E-07
W-187	1.8E-06	1.8E-06
Tritium		
H-3	1.0E-03	4.00E+00[3]

Notes:

1. Concentrations for all nuclides except halogens, activation products, and tritium, based on a conservative fuel defect level of 0.25% failed fuel fraction.
2. Concentrations for all nuclides except halogens, activation products, and tritium, based on a conservative fuel defect level of 1.0% failed fuel fraction.
3. Conservatively assumed to be same concentration as in primary coolant.

Table 11.1-5—Secondary Coolant Design Basis Source Term, Steam Concentrations
Sheet 1 of 4

Nuclide	Secondary Steam Concentration, Shielding (µCi/g)[1]	Secondary Steam Concentration, Effluent (µCi/g)[2]
Noble Gases		
Kr-83m	2.1E-05	2.1E-05
Kr-85m	5.8E-06	5.8E-06
Kr-85	5.4E-05	5.3E-05
Kr-87	3.3E-06	3.3E-06
Kr-88	1.0E-05	1.0E-05
Kr-89	2.4E-07	2.4E-07
Xe-131m	1.1E-05	1.1E-05
Xe-133m	1.5E-05	1.5E-05
Xe-133	9.7E-04	9.7E-04
Xe-135m	8.2E-04	8.2E-04
Xe-135	1.6E-04	1.6E-04
Xe-137	4.6E-07	4.6E-07
Xe-138	1.6E-06	1.7E-06
Halogens		
Br-83	1.6E-05	1.6E-05
Br-84	3.1E-06	3.1E-06
Br-85	3.9E-08	3.9E-08
I-129	4.8E-11	4.8E-11
I-130	4.3E-05	4.3E-05
I-131	7.7E-04	7.7E-04
I-132	2.3E-04	2.3E-04
I-133	1.2E-03	1.2E-03
I-134	6.7E-05	6.7E-05
I-135	6.0E-04	6.0E-04
Rubidium, Cesium		
Rb-86m	1.1E-14	4.5E-14
Rb-86	1.9E-08	7.7E-08
Rb-88	6.3E-07	2.5E-06
Rb-89	2.5E-08	1.0E-07

**Table 11.1-5—Secondary Coolant Design Basis Source Term, Steam Concentrations
Sheet 2 of 4**

Nuclide	Secondary Steam Concentration, Shielding (µCi/g)[1]	Secondary Steam Concentration, Effluent (µCi/g)[2]
Cs-134	1.7E-06	6.9E-06
Cs-136	5.2E-07	2.1E-06
Cs-137	1.1E-06	4.3E-06
Cs-138	2.3E-07	9.4E-07
Miscellaneous Nuclides		
Sr-89	3.6E-09	1.4E-08
Sr-90	1.8E-10	7.2E-10
Sr-91	4.5E-09	1.8E-08
Sr-92	5.0E-10	2.0E-09
Y-90	4.8E-11	1.9E-10
Y-91m	2.7E-09	1.1E-08
Y-91	4.6E-10	1.8E-09
Y-92	6.7E-10	2.7E-09
Y-93	2.9E-10	1.2E-09
Zr-95	5.2E-10	2.1E-09
Zr-97	3.2E-10	1.3E-09
Nb-95	5.2E-10	2.1E-09
Mo-99	5.7E-07	2.3E-06
Tc-99m	3.3E-07	1.3E-06
Ru-103	4.3E-10	1.7E-09
Ru-105	3.5E-10	1.4E-09
Ru-106	1.5E-10	6.0E-10
Rh-103m	3.9E-10	1.5E-09
Rh-105	2.5E-10	1.0E-09
Rh-106	1.5E-10	6.0E-10
Ag-110m	1.1E-12	4.4E-12
Ag-110	1.5E-14	5.9E-14
Sb-125	4.4E-12	1.8E-11
Sb-127	2.7E-11	1.1E-10
Sb-129	2.4E-11	9.6E-11
Te-127m	2.4E-09	9.8E-09

Table 11.1-5—Secondary Coolant Design Basis Source Term, Steam Concentrations
Sheet 3 of 4

Nuclide	Secondary Steam Concentration, Shielding (µCi/g)[1]	Secondary Steam Concentration, Effluent (µCi/g)[2]
Te-127	1.0E-08	4.0E-08
Te-129m	8.1E-09	3.2E-08
Te-129	7.8E-09	3.1E-08
Te-131m	1.9E-08	7.7E-08
Te-131	5.7E-09	2.3E-08
Te-132	2.2E-07	8.8E-07
Te-134	8.1E-09	3.2E-08
Ba-137m	1.0E-06	4.1E-06
Ba-139	4.8E-08	1.9E-07
Ba-140	3.4E-09	1.4E-08
La-140	1.1E-09	4.2E-09
La-141	1.9E-10	7.4E-10
La-142	7.0E-11	2.8E-10
Ce-141	4.9E-10	2.0E-09
Ce-143	3.9E-10	1.6E-09
Ce-144	3.9E-10	1.5E-09
Pr-143	4.9E-10	2.0E-09
Pr-144	3.9E-10	1.5E-09
Nd-147	1.9E-10	7.5E-10
Np-239	4.7E-09	1.9E-08
Pu-238	1.1E-12	4.4E-12
Pu-239	1.1E-13	4.5E-13
Pu-240	1.6E-13	6.2E-13
Pu-241	3.8E-11	1.5E-10
Am-241	4.4E-14	1.7E-13
Cm-242	1.0E-11	4.2E-11
Cm-244	5.7E-13	2.3E-12
Activation Products		
Na-24	1.8E-07	1.8E-07
Cr-51	1.1E-08	1.1E-08
Mn-54	5.6E-09	5.6E-09

Table 11.1-5—Secondary Coolant Design Basis Source Term, Steam Concentrations
Sheet 4 of 4

Nuclide	Secondary Steam Concentration, Shielding (µCi/g)[1]	Secondary Steam Concentration, Effluent (µCi/g)[2]
Fe-55	4.2E-09	4.2E-09
Fe-59	1.1E-09	1.1E-09
Co-58	1.6E-08	1.6E-08
Co-60	1.9E-09	1.9E-09
Zn-65	1.8E-09	1.8E-09
W-187	9.1E-09	9.1E-09
Tritium		
H-3	1.0E-03	4.0E+00[3]

Notes:

1. Concentrations for all nuclides except noble gases, halogens, activation products, and tritium, based on a conservative fuel defect level of 0.25% failed fuel fraction.

2. Concentrations for all nuclides except noble gases, halogens, activation products, and tritium, based on a conservative fuel defect level of 1.0% failed fuel fraction.

3. Conservatively assumed to be the same concentration as in secondary liquid.

Table 11.1-6—Parameters Used to Calculate Realistic Source Terms

Parameter	Element Class	Value	
		U.S. EPR	ANS/ANSI-18.1-1999 Reference Plant
Thermal power (MW_t)	N/A	4612	3400
Steam flow rate (lb_m/hr)	N/A	1.9E+07	1.5E+07
Mass of water in RCS (lb_m)	N/A	6.47E+05	5.5E+05
Mass of water in SGs (lb_m)	N/A	6.5E+05	4.5E+05
RCS letdown purification flow rate (lb_m/hr)	N/A	7.94E+04	3.7E+04
RCS letdown flow rate for boron control (lb_m/hr)	N/A	500	500
Total SG blowdown flow rate (lb_m/hr)	N/A	1.9E+05	7.5E+04
Fraction of radioactivity in blowdown steam which is not returned to the secondary coolant system	N/A	50%	100%
Purification system cation demineralizer flow (lb_m/hr)	N/A	0	3.7E+03
Ratio of condensate demineralizer flow rate to total steam flow	N/A	0.0	0.0
Fraction of noble gas activity in letdown stream that is not returned to the RCS (excluding the boron recovery system)	N/A	0.0	0.0
Fraction of material removed in passing through the cation demineralizers	1, 2, 4, 5	0.0	0.0
	3, 6	0.0	0.9
Fraction of material removed in passing through the purification demineralizers	1, 4, 5	0.0	0.0
	2	0.99	0.99
	3	0.5	0.5
	6	0.98	0.98
Fraction of activity removed in passing through the condensate demineralizers	1, 4, 5	0.0	0.0
	2, 6	0.9	0.9
	3	0.5	0.5
Ratio of concentration in steam to that in the steam generator (U-tube design)	1, 4	N/A	N/A
	2	0.01	0.01
	3, 6,	0.005	0.005
	5	1.0	1.0
Primary to secondary leak rate (lb_m/day)	N/A	75	75

Table 11.1-7—RCS and Secondary Coolant System Realistic Source Terms
Sheet 1 of 3

ANSI/ANS Element Class	Radionuclide	Concentrations (µCi/gm)		
		RCS	Secondary Coolant Water	Secondary Coolant Steam
Class 1 (Noble Gases)	Kr-85m	1.8E-02	0.0E+00	3.1E-09
	Kr-85	5.8E-01	0.0E+00	9.5E-08
	Kr-87	2.0E-02	0.0E+00	9.1E-09
	Kr-88	2.1E-02	0.0E+00	3.5E-09
	Xe-131m	8.8E-01	0.0E+00	1.4E-07
	Xe-133m	8.2E-02	0.0E+00	1.4E-08
	Xe-133	3.4E-02	0.0E+00	5.6E-09
	Xe-135m	1.5E-01	0.0E+00	2.5E-08
	Xe-135	7.7E-02	0.0E+00	1.3E-08
	Xe-137	3.9E-02	0.0E+00	6.5E-09
	Xe-138	7.0E-02	0.0E+00	1.2E-08
	Subtotal	2.0E+00	0.0E+00	3.3E-07
Class 2 (Halogens)	Br-84	1.8E-02	5.8E-08	5.8E-10
	I-131	1.3E-03	4.1E-08	4.1E-10
	I-132	6.0E-02	6.5E-07	6.5E-09
	I-133	1.9E-02	5.2E-07	5.2E-09
	I-134	1.1E-01	5.5E-07	5.5E-09
	I-135	4.8E-02	9.2E-07	9.2E-09
	Subtotal	2.6E-01	2.7E-06	2.7E-08
Class 3 (Rubidium/ Cesium)	Rb-88	2.2E-01	4.2E-07	2.1E-09
	Cs-134	2.6E-05	9.3E-10	4.9E-12
	Cs-136	6.2E-04	2.2E-08	1.1E-10
	Cs-137	3.7E-05	1.4E-09	6.6E-12
	Subtotal	2.2E-01	4.5E-07	2.2E-09
Class 4	N-16	4.0E+01	6.9E-07	6.9E-08
Class 5	H-3	1.0E+00	1.0E-03	1.0E-03
Class 6 (Miscellaneous Nuclides)	Na-24	3.7E-02	8.9E-07	4.5E-09
	Cr-51	2.0E-03	6.5E-08	3.2E-10
	Mn-54	1.0E-03	3.3E-08	1.7E-10
	Fe-55	7.6E-04	2.5E-08	1.3E-10

Table 11.1-7—RCS and Secondary Coolant System Realistic Source Terms
Sheet 2 of 3

ANSI/ANS Element Class	Radionuclide	Concentrations (µCi/gm)		
		RCS	Secondary Coolant Water	Secondary Coolant Steam
Class 6 (Miscellaneous Nuclides)	Fe-59	1.9E-04	6.0E-09	3.1E-11
	Co-58	2.9E-03	9.5E-08	4.7E-10
	Co-60	3.4E-04	1.1E-08	5.5E-11
	Zn-65	3.2E-04	1.1E-08	5.0E-11
	Sr-89	8.9E-05	2.9E-09	1.5E-11
	Sr-90	7.6E-06	2.5E-10	1.2E-12
	Sr-91	8.0E-04	1.8E-08	8.8E-11
	Y-91m	5.0E-04	2.5E-09	1.2E-11
	Y-91	3.3E-06	1.1E-10	5.5E-13
	Y-93	3.5E-03	7.5E-08	3.8E-10
	Zr-95	2.5E-04	8.0E-09	4.0E-11
	Nb-95	1.8E-04	5.5E-09	2.9E-11
	Mo-99	4.3E-03	1.3E-07	6.3E-10
	Tc-99m	4.2E-03	7.3E-08	3.8E-10
	Ru-103	4.8E-03	1.6E-07	8.0E-10
	Ru-106	5.7E-02	1.9E-06	9.0E-09
	Ag-110m	8.2E-04	2.7E-08	1.4E-10
	Te-129m	1.2E-04	3.9E-09	2.0E-11
	Te-129	2.6E-02	1.7E-07	8.3E-10
	Te-131m	1.1E-03	3.0E-08	1.5E-10
	Te-131	8.6E-03	2.3E-08	1.2E-10
	Te-132	1.1E-03	3.5E-08	1.7E-10
	Ba-140	8.4E-03	2.6E-07	1.3E-09
	La-140	1.7E-02	5.1E-07	2.5E-09
	Ce-141	9.6E-05	3.1E-09	1.6E-11
	Ce-143	2.0E-03	5.5E-08	2.8E-10
	Ce-144	2.5E-03	8.0E-08	4.1E-10
	W-187	1.8E-03	4.9E-08	2.5E-10
	Np-239	1.5E-03	4.5E-08	2.2E-10
	Subtotal	2.0E-01	4.9E-06	2.5E-08

Table 11.1-7—RCS and Secondary Coolant System Realistic Source Terms
Sheet 3 of 3

ANSI/ANS Element Class	Radionuclide	Concentrations (µCi/gm) RCS	Secondary Coolant Water	Secondary Coolant Steam
Nuclides in Secular Equilibrium	Ba-137m Cs-137*0.946 (branching fraction)	3.5E-05	1.3E-09	6.2E-12
	Rh-106 (Ru-106*1.0)	5.7E-02	1.9E-06	9.0E-09
	Pr-144 (Ce-144*1.0)	2.5E-03	8.0E-08	4.1E-10
	Subtotal	6.0E-02	1.9E-06	9.4E-09

Next File