REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**238-8145**
SRP Section:	**11.01 – Source Terms**
Application Section:	**11.1 SRP 11.1**
Date of RAI Issue:	**10/08/2015**

Question No. 11.01-4

Staff requests clarification regarding information outside the normal operating source term provided in DCD Section 11.1. SRP Section 11.1 only instructs the review of the normal operation source term including anticipated operational occurrences. There is information contained in DCD Section 11.1 that describes the source terms for other sections in the DCD that are not referenced within DCD Section 11.1.

Please clarify the DCD text to direct staff on the use of equations and tables in the DCD. These clarifications should direct the staff to other DCD sections in which the referenced source terms are used, and provide direct footnotes in the tables located in DCD Section 11.1. The DCD text of 11.1 should also discuss the use of the tables provided in DCD section 11.1

Please address the items above and provide a mark-up on the proposed DCD changes

Response – (Rev. 2)

DCD section 11.1 will be updated to direct the staff where source terms are used in other DCD sections.

Impact on DCD

DCD section 11.1 will be revised as indicated in the Attachment.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical or Environmental Report.

11.1.3 Neutron Activation Products

11.1.3.1 Deposited Crud Activities

The half-lives, reactions, and gamma decay energies for each of the long-lived isotopes in the radioactive crud are used to calculate the activities of deposited crud and provided in Table 11.1-11.

Deposited crud activities on primary system surfaces have been evaluated using measured data from various operating pressurized water reactors (PWRs). Even though these reactors have different water chemistries and different materials in contact with the primary coolant, their crud activities (μCi/g-crud (Bq/g-crud)) ~~(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:

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

Deposited crud:

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

Where:

A_i, A_j = crud activities for each isotope, μCi/g-crud (Bq/g-crud) ~~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 used to calculate the activities of deposited crud and shown in Table 11.1-12.

(μCi/g-crud (Bq/g-crud))

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.

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

μCi/g-crud (Bq/g-crud)

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.

(μCi/g-crud (Bq/g-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

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

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

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

(μCi (Bq))

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

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.

and can be used to check the adequacy of the tritium activities in Tables 11.1-2 and 11.1-9.

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 SG 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.

This table should be replaced with table on page 5 of Attachment.

Table 11.1-2

Maximum Reactor Coolant Fission Product Source Term (5)
(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) Expected source terms based on ANSI/ANS 18.1 (Reference 1) are used when these values are higher than the design basis source terms, for added conservatism.

(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

(5) This source term is used to determine the design basis radioactive source terms for LWMS tanks, pumps, and other miscellaneous components listed in Table 11.2-13 and Table 11.2-14.

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Nuclide	Specific Activity	
	μCi/g	Bq/g
Kr-85m	8.10E-01	3.00E+04
Kr-85	2.00E-02	7.40E+02
Kr-87	7.90E-01	2.92E+04
Kr-88	2.00E+00	7.40E+04
Xe-131m	2.00E-01	7.40E+03
Xe-133m	5.20E-02	1.92E+03
Xe-133	2.60E+01	9.62E+05
Xe-135m	6.20E-01	2.29E+04
Xe-135	3.50E+00	1.30E+05
Xe-137	1.50E-01	5.55E+03
Xe-138	5.30E-01	1.96E+04
Br-84	2.10E-02	7.77E+02
I-131	2.70E+00	9.99E+04
I-132	7.20E-01	2.66E+04
I-133	3.80E+00	1.41E+05
I-134	4.50E-01	1.67E+04
I-135	2.10E+00	7.77E+04
Rb-88	2.00E+00	7.40E+04
Cs-134	3.80E-01	1.41E+04
Cs-136	5.10E-02	1.89E+03
Cs-137	4.40E-01	1.63E+04
N-16	2.22E+02(3	8.22E+06(3)
H-3	3.51E+00(4)	1.30E+05(4)
Na-24	4.89E-02(1)	1.81E+03(1)
Cr-51	1.48E-02	5.48E+02
Mn-54	1.71E-03(1)	6.34E+01(1)
Fe-55	1.28E-03(1)	4.75E+01(1)
Fe-59	3.21E-04(1)	1.19E+01(1)
Co-58	4.92E-03(1)	1.82E+02(1)

Nuclide	Specific Activity	
	μCi/g	Bq/g
Co-60	5.67E-04(1)	2.10E+01(1)
Zn-65	5.46E-04(1)	2.02E+01(1)
Sr-89	3.50E-03	1.30E+02
Sr-90	2.40E-04	8.88E+00
Sr-91	5.20E-03	1.92E+02
Y-91m	3.00E-03	1.11E+02
Y-91	5.10E-04	1.89E+01
Y-93	1.20E-04	4.44E+00
Zr-95	6.49E-04(2)	2.40E+01(2)
Nb-95	5.50E-04	2.04E+01
Mo-99	3.00E-01	1.11E+04
Tc-99m	1.80E-01	6.66E+03
Ru-103	1.90E-04	7.03E+00
Ru-106	8.10E-05	3.00E+00
Ag-110m	1.39E-03(1)	5.15E+01(1)
Te-129m	6.40E-03	2.37E+02
Te-129	6.80E-03	2.52E+02
Te-131m	3.00E-02	1.11E+03
Te-131	1.20E-02	4.44E+02
Te-132	2.10E-01	7.77E+03
Ba-137m	4.20E-01	1.55E+04
Ba-140	4.30E-03	1.59E+02
La-140	1.50E-03	5.55E+01
Ce-141	1.60E-04	5.92E+00
Ce-143	4.50E-04	1.67E+01
Ce-144	4.60E-04	1.70E+01
W-187	2.62E-03(1)	9.70E+01(1)
Np-239	2.33E-03(1)	8.62E+01(1)

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This table should be replaced with table on page 7 of Attachment.

Table 11.1-6

(1)

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

Nuclide	Steam Generator Liquid	Steam Generator 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 Generator 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

(1) This source term is used to determine the design basis radioactive source terms for SGBD components.

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Nuclide	Steam Generator			
	Liquid		Steam	
	(μCi/g)	(Bq/g)	(μCi/g)	(Bq/g)
Kr-85m	-	-	1.00E-05	3.71E-01
Kr-85	-	-	2.48E-07	9.16E-03
Kr-87	-	-	9.76E-06	3.61E-01
Kr-88	-	-	2.48E-05	9.16E-01
Xe-131m	-	-	2.48E-06	9.16E-02
Xe-133m	-	-	6.43E-07	2.38E-02
Xe-133	-	-	3.22E-04	1.19E+01
Xe-135m	-	-	7.65E-06	2.83E-01
Xe-135	-	-	4.35E-05	1.61E+00
Xe-137	-	-	1.86E-06	6.87E-02
Xe-138	-	-	6.57E-06	2.43E-01
Br-84	8.19E-06	3.03E-01	8.19E-08	3.03E-03
I-131	3.24E-03	1.20E+02	3.24E-05	1.20E+00
I-132	5.86E-04	2.17E+01	5.86E-06	2.17E-01
I-133	4.38E-03	1.62E+02	4.38E-05	1.62E+00
I-134	2.39E-04	8.86E+00	2.39E-06	8.86E-02
I-135	2.18E-03	8.08E+01	2.18E-05	8.08E-01
Rb-88	5.22E-04	1.93E+01	2.61E-06	9.64E-02
Cs-134	5.16E-04	1.91E+01	2.58E-06	9.56E-02
Cs-136	6.89E-05	2.55E+00	3.46E-07	1.28E-02
Cs-137	5.97E-04	2.21E+01	3.00E-06	1.11E-01
Cr-51	1.83E-05	6.78E-01	9.16E-08	3.39E-03
Mn-54	2.12E-06	7.85E-02	1.06E-08	3.93E-04
Fe-55	1.59E-06	5.88E-02	7.95E-09	2.94E-04
Fe-59	3.97E-07	1.47E-02	1.99E-09	7.36E-05
Co-58	6.08E-06	2.25E-01	3.05E-08	1.13E-03
Co-60	7.03E-07	2.60E-02	3.51E-09	1.30E-04
Zr-95	8.03E-07	2.97E-02	4.03E-09	1.49E-04
Zn-65	6.76E-07	2.50E-02	3.38E-09	1.25E-04

Nuclide	Steam Generator			
	Liquid		Steam	
	(μCi/g)	(Bq/g)	(μCi/g)	(Bq/g)
N-16	1.99E-05	7.38E-01	9.97E-08	3.69E-03
Na-24	5.62E-05	2.08E+00	2.81E-07	1.04E-02
Sr-89	4.35E-06	1.61E-01	2.17E-08	8.04E-04
Sr-90	2.97E-07	1.10E-02	1.49E-09	5.50E-05
SR-91	5.76E-06	2.13E-01	2.86E-08	1.06E-03
Y-91m	1.57E-06	5.81E-02	7.86E-09	2.91E-04
Y-91	6.32E-07	2.34E-02	3.16E-09	1.17E-04
Y-93	1.34E-07	4.95E-03	6.68E-10	2.47E-05
Nb-95	6.81E-07	2.52E-02	3.41E-09	1.26E-04
Mo-99	3.65E-04	1.35E+01	1.83E-06	6.76E-02
Tc-99M	1.87E-04	6.93E+00	9.35E-07	3.46E-02
Ru-103	2.35E-07	8.70E-03	1.18E-09	4.35E-05
Ru-106	1.01E-07	3.72E-03	5.03E-10	1.86E-05
Ag-110m	1.72E-06	6.38E-02	8.62E-09	3.19E-04
Te-129m	7.92E-06	2.93E-01	3.97E-08	1.47E-03
Te-129	4.24E-06	1.57E-01	2.12E-08	7.84E-04
Te-131m	3.57E-05	1.32E+00	1.79E-07	6.62E-03
Te-131	4.00E-06	1.48E-01	1.99E-08	7.38E-04
Te-132	2.56E-04	9.49E+00	1.28E-06	4.74E-02
Ba-137m	1.87E-05	6.92E-01	9.35E-08	3.46E-03
Ba-140	5.30E-06	1.96E-01	2.65E-08	9.81E-04
La-140	1.81E-06	6.69E-02	9.03E-09	3.34E-04
Ce-141	1.98E-07	7.32E-03	9.89E-10	3.66E-05
Ce-143	5.41E-07	2.00E-02	2.70E-09	1.00E-04
Ce-144	5.70E-07	2.11E-02	2.84E-09	1.05E-04
W-187	3.11E-06	1.15E-01	1.55E-08	5.73E-04
Np-239	2.84E-06	1.05E-01	1.41E-08	5.23E-04
H-3	4.57E-01	1.69E+04	4.57E-01	1.69E+04

Table 11.1-7

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

(1)

Delete

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

This table should be replaced with table on page 9 of Attachment.

(1) The crud concentrations in this Table is presented for information only.

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Nuclide	Design Basis Source		Expected Source	
	(μCi/g)	(Bq/g)	(μCi/g)	(Bq/g)
Cr-51	8.81E-03	3.26E+02	8.43E-04	3.12E+01
Mn-54	6.57E-01	2.43E+04	7.43E-05	2.75E+00
Fe-55	7.38E-02	2.73E+03	1.45E-03	5.35E+01
Fe-59	7.62E-01	2.82E+04	1.06E-04	3.93E+00
Co-58	2.35E-02	8.71E+02	5.27E-03	1.95E+02
Co-60	2.95E-03	1.09E+02	2.95E-03	1.09E+02
Zr-95	2.22E-03	8.21E+01	2.22E-03	8.21E+01
Zn-65	5.27E-04	1.95E+01	5.27E-04	1.95E+01

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This table should be replaced with table on page 11 of Attachment.

Table 11.1-8

Design Basis Radionuclide Concentrations of Sources to GRS (Bq/cm^3)[(1)] Delete

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 GRS.

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

This source term is used to determine the design basis radioactive source terms for GRS components listed in Table 11.3-11

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Nuclide	Reactor Drain Tank(2)		Volume Control Tank		Gas Stripper(2)		Equipment Drain Tank(3)	
	μCi/cm3	Bq/cm3	μCi/cm3	Bq/cm3	μCi/cm3	Bq/cm3	μCi/cm3	Bq/cm3
H-3	2.8E-04	1.0E+01	2.8E-04	1.0E+01	4.6E-04	1.7E+01	2.8E-05	1.0E+00
Br-84	2.1E-05	7.6E-01	2.8E-07	1.0E-02	7.0E-06	2.6E-01	1.9E-07	7.2E-03
Kr-85m	7.2E-01	2.7E+04	2.8E-02	1.0E+03	2.7E+01	1.0E+06	6.8E-02	2.5E+03
Kr-85	1.8E-02	6.6E+02	7.4E-04	2.7E+01	6.7E-01	2.5E+04	1.7E-03	6.2E+01
Kr-87	7.0E-01	2.6E+04	2.4E-02	8.9E+02	2.6E+01	9.8E+05	6.6E-02	2.5E+03
Kr-88	1.8E+00	6.6E+04	6.6E-02	2.5E+03	6.7E+01	2.5E+06	1.7E-01	6.2E+03
Xe-131m	1.8E-01	6.6E+03	4.1E-03	1.5E+02	6.7E+00	2.5E+05	1.7E-02	6.2E+02
Xe-133m	4.6E-02	1.7E+03	1.0E-03	3.9E+01	1.7E+00	6.4E+04	4.4E-03	1.6E+02
Xe-133	2.3E+01	8.6E+05	5.4E-01	2.0E+04	8.7E+02	3.2E+07	2.2E+00	8.1E+04
Xe-135m	5.5E-01	2.0E+04	6.6E-03	2.5E+02	2.1E+01	7.7E+05	5.2E-02	1.9E+03
Xe-135	3.1E+00	1.2E+05	6.9E-02	2.5E+03	1.2E+02	4.4E+06	3.0E-01	1.1E+04
Xe-137	1.3E-01	4.9E+03	7.4E-04	2.7E+01	5.0E+00	1.9E+05	1.3E-02	4.7E+02
Xe-138	4.7E-01	1.7E+04	5.4E-03	2.0E+02	1.8E+01	6.6E+05	4.5E-02	1.7E+03
I-131	2.6E-03	9.8E+01	4.5E-05	1.7E+00	9.0E-04	3.3E+01	2.5E-05	9.3E-01
I-132	7.0E-04	2.6E+01	1.1E-05	4.2E-01	2.4E-04	8.9E+00	6.7E-06	2.5E-01
I-133	3.7E-03	1.4E+02	6.2E-05	2.3E+00	1.3E-03	4.7E+01	3.5E-05	1.3E+00
I-134	4.4E-04	1.6E+01	6.5E-06	2.4E-01	1.5E-04	5.6E+00	4.2E-06	1.5E-01
I-135	2.1E-03	7.6E+01	3.4E-05	1.2E+00	7.0E-04	2.6E+01	1.9E-05	7.2E-01

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This table should be replaced with table on page 13 of Attachment.

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).

. This source term is used to determine the source terms for the expected radioactive source terms for LWMS tanks and other components listed in Table 11.2-11 and Table 11.2-12.

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Nuclide	Specific Activity	
	μCi/g	Bq/g
Kr-85m	1.61E-02	5.96E+02
Kr-85	1.17E+00	4.33E+04
Kr-87	1.71E-02	6.32E+02
Kr-88	1.81E-02	6.70E+02
Xe-131m	8.85E-01	3.27E+04
Xe-133m	7.32E-02	2.71E+03
Xe-133	3.20E-02	1.18E+03
Xe-135m	1.31E-01	4.83E+03
Xe-135	6.78E-02	2.51E+03
Xe-137	3.41E-02	1.26E+03
Xe-138	6.12E-02	2.27E+03
Br-84	1.61E-02	5.97E+02
I-131	2.22E-03	8.23E+01
I-132	6.14E-02	2.27E+03
I-133	2.80E-02	1.04E+03
I-134	1.01E-01	3.74E+03
I-135	5.75E-02	2.13E+03
Rb-88	1.91E-01	7.07E+03
Cs-134	4.29E-05	1.59E+00
Cs-136	1.00E-03	3.70E+01
Cs-137	6.15E-05	2.27E+00
N-16	4.00E+01(2)	1.48E+06(2)
H-3	1.00E+00(3)	3.70E+04(3)
Na-24	4.89E-02	1.81E+03
Cr-51	3.32E-03	1.23E+02
Mn-54	1.71E-03	6.34E+01
Fe-55	1.28E-03	4.75E+01
Fe-59	3.21E-04	1.19E+01

Nuclide	Specific Activity	
	μCi/g	Bq/g
Co-58	4.92E-03	1.82E+02
Co-60	5.67E-04	2.10E+01
Zn-65	5.46E-04	2.02E+01
Sr-89	1.50E-04	5.54E+00
Sr-90	1.28E-05	4.75E-01
Sr-91	9.93E-04	3.67E+01
Y-91m	4.64E-04	1.72E+01
Y-91	5.57E-06	2.06E-01
Y-93	4.35E-03	1.61E+02
Zr-95	4.17E-04	1.54E+01
Nb-95	3.00E-04	1.11E+01
Mo-99	6.79E-03	2.51E+02
Tc-99m	4.83E-03	1.79E+02
Ru-103	8.02E-03	2.97E+02
Ru-106	9.64E-02	3.57E+03
Ag-110m	1.39E-03	5.15E+01
Te-129m	2.03E-04	7.52E+00
Te-129	2.42E-02	8.97E+02
Te-131m	1.58E-03	5.84E+01
Te-131	7.75E-03	2.87E+02
Te-132	1.81E-03	6.68E+01
Ba-137m	6.15E-05	2.27E+00
Ba-140	1.39E-02	5.14E+02
La-140	2.64E-02	9.77E+02
Ce-141	1.60E-04	5.94E+00
Ce-143	2.95E-03	1.09E+02
Ce-144	4.28E-03	1.58E+02
W-187	2.62E-03	9.70E+01
Np-239	2.33E-03	8.62E+01

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This table should be replaced with table on page 15 of Attachment.

Table 11.1-10

(1)

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

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

(1) This source term is used to determine the expected radioactive source terms for SGBD components

and this source terms are presented for information only.

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Nuclide	Steam Generator			
	Liquid		Steam	
	(μCi/g)	(Bq/g)	(μCi/g)	(Bq/g)
Kr-85M	-	-	2.81E-09	1.04E-04
Kr-85	-	-	2.04E-07	7.54E-03
Kr-87	-	-	2.97E-09	1.10E-04
Kr-88	-	-	3.16E-09	1.17E-04
Xe-131m	-	-	1.54E-07	5.69E-03
Xe-133 m	-	-	1.28E-08	4.72E-04
Xe-133	-	-	5.54E-09	2.05E-04
Xe-135 m	-	-	2.27E-08	8.41E-04
Xe-135	-	-	1.18E-08	4.37E-04
Xe-137	-	-	5.92E-09	2.19E-04
Xe-138	-	-	1.07E-08	3.95E-04
Br-84	8.86E-08	3.28E-03	8.86E-10	3.28E-05
I-131	3.76E-08	1.39E-03	3.76E-10	1.39E-05
I-132	7.05E-07	2.61E-02	7.05E-09	2.61E-04
I-133	4.54E-07	1.68E-02	4.54E-09	1.68E-04
I-134	7.54E-07	2.79E-02	7.54E-09	2.79E-04
I-135	8.43E-07	3.12E-02	8.43E-09	3.12E-04
Rb-88	7.00E-07	2.59E-02	3.51E-09	1.30E-04
Cs-134	8.19E-10	3.03E-05	4.11E-12	1.52E-07
Cs-136	1.90E-08	7.03E-04	9.51E-11	3.52E-06
Cs-137	1.17E-09	4.33E-05	5.86E-12	2.17E-07
Cr-51	5.78E-08	2.14E-03	2.89E-10	1.07E-05
Mn-54	2.97E-08	1.10E-03	1.49E-10	5.52E-06
Fe-55	2.24E-08	8.28E-04	1.12E-10	4.14E-06
Fe-59	5.59E-09	2.07E-04	2.81E-11	1.04E-06
Co-58	8.57E-08	3.17E-03	4.27E-10	1.58E-05
Co-60	9.89E-09	3.66E-04	4.95E-11	1.83E-06
Zr-95	7.24E-09	2.68E-04	3.62E-11	1.34E-06
Zn-65	9.51E-09	3.52E-04	4.76E-11	1.76E-06

Nuclide	Steam Generator			
	Liquid		Steam	
	(μCi/g)	(Bq/g)	(μCi/g)	(Bq/g)
N-16	5.05E-08	1.87E-03	2.53E-10	9.35E-06
Na-24	7.92E-07	2.93E-02	3.97E-09	1.47E-04
Sr-89	2.61E-09	9.65E-05	1.30E-11	4.82E-07
Sr-90	2.24E-10	8.28E-06	1.12E-12	4.14E-08
Sr-91	1.55E-08	5.72E-04	7.73E-11	2.86E-06
Y-91m	3.43E-09	1.27E-04	1.71E-11	6.33E-07
Y-91	9.70E-11	3.59E-06	4.84E-13	1.79E-08
Y-93	6.81E-08	2.52E-03	3.41E-10	1.26E-05
Nb-95	5.22E-09	1.93E-04	2.61E-11	9.66E-07
Mo-99	1.16E-07	4.30E-03	5.81E-10	2.15E-05
Tc-99m	7.08E-08	2.62E-03	3.54E-10	1.31E-05
Ru-103	1.40E-07	5.17E-03	6.97E-10	2.58E-05
Ru-106	1.68E-06	6.22E-02	8.41E-09	3.11E-04
Ag-110m	2.42E-08	8.97E-04	1.21E-10	4.49E-06
Te-129m	3.54E-09	1.31E-04	1.77E-11	6.54E-07
Te-129	2.12E-07	7.86E-03	1.06E-09	3.93E-05
Te-131m	2.65E-08	9.81E-04	1.32E-10	4.90E-06
Te-131	3.62E-08	1.34E-03	1.81E-10	6.71E-06
Te-132	3.11E-08	1.15E-03	1.55E-10	5.74E-06
Ba-137m	3.86E-11	1.43E-06	1.93E-13	7.13E-09
Ba-140	2.41E-07	8.92E-03	1.21E-09	4.46E-05
La-140	4.49E-07	1.66E-02	2.24E-09	8.28E-05
Ce-141	2.78E-09	1.03E-04	1.40E-11	5.17E-07
Ce-143	4.97E-08	1.84E-03	2.48E-10	9.18E-06
Ce-144	7.43E-08	2.75E-03	3.73E-10	1.38E-05
W-187	4.35E-08	1.61E-03	2.18E-10	8.07E-06
Np-239	3.97E-08	1.47E-03	1.99E-10	7.36E-06
H-3	1.84E-03	6.81E+01	1.84E-03	6.81E+01

Table 11.1-11

(2)

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

(2) The half-lives, reactions, and gamma decay energies for each of the long-lived isotopes in the radioactive crud are used to calculate the activities of deposited crud in section 11.1.3.1.

Table 11.1-12

(1)

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

(1)

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

(1) The averages (t_{res}) of the maximum core residence times, the system parameters, and the activation rates are used to determine crud activities in section 11.1.3.1.

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Table 11.1-13

Long-Lived Crud Activity (1)

This table should be replaced with table on page 19 of Attachment.

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

(1) These calculated crud activities are applied to both the circulating crud and out-of-core deposited crud in section 11.1.3.1

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Isotope	Half-life	Activity	
		μCi/g-crud	Bq/g-crud
Cr-51	27.70 days	1.98E+05	7.31E+09
Mn-54	312.3 days	5.37E+02	1.99E+07
Fe-59	44.50 days	1.13E+03	4.18E+07
Co-58	70.82 days	4.78E+04	1.77E+09
Co-60	5.272 years	1.68E+03	6.22E+07
Zr-95	64.02 days	1.32E+03	4.90E+07

Table 11.1-14

Calculated Average Crud Activity
in the Reactor Coolant (1)

This table should be replaced with table on page 21 of Attachment.

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

(1) The crud activity in this table is presented for information only.

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Isotope	Activity	
	μCi/g-coolant	Bq/g-coolant
Cr-51	1.48E-02	5.48E+02
Mn-54	4.03E-05	1.49E+00
Fe-59	8.48E-05	3.14E+00
Co-58	3.58E-03	1.33E+02
Co-60	1.26E-04	4.66E+00
Zr-95	9.92E-05	3.67E+00

Table 11.1-15

Tritium Activation Reactions (2)

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

(2) The activation reactions from B-10, lithium, and deuterium are used to calculate tritium production in Table 11.1-17.

Table 11.1-16

Parameters Used for Calculating Tritium Production (1)

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

(1) The parameters are used to calculate tritium production in Table 11.1-17.

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This table should be replaced with table on page 25 of Attachment.

Table 11.1-17

Tritium Production in the Reactor Coolant (2)

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)

(2) The tritium produced from activation sources are used to check the adequacy of the tritium activities in Tables 11.1-2 and 11.1-9.

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Source	Average		Maximum	
	μCi/cycle	Bq/cycle	μCi/cycle	Bq/cycle
H^2(n, γ)T	6.95E+06	2.57E+11	6.95E+06	2.57E+11
Li^6(n, α)T	2.94E+08	1.09E+13	4.67E+08	1.73E+13
Li^7(n, nα)T	1.86E+07	6.90E+11	2.97E+07	1.10E+12
B^{10}(n, 2α)T	1.16E+09	4.28E+13	1.34E+09	4.95E+13
Fission products(1)	2.80E+08	1.04E+13	5.60E+08	2.07E+13
CEAs	6.23E+07	2.30E+12	3.49E+08	1.29E+13
Total	1.82E+09	6.73E+13	2.75E+09	1.02E+14

Table 11.1-18

Maximum Anticipated Leakage Rates from NSSS-Related Components to the Building Environment (1)

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

(1) These assumed leakage rates are used to determine the airborne concentration described in section 12.2.2.3.