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.: **10-7850**

SRP Section: **11.01 – Source Terms**

Application Section: **SRP 11.1**

Date of RAI Issued: **05/14/2015**

Question No. 11.01-1

The Standard Review Plan (SRP) 11.1 utilizes various source terms for a variety of purposes, including: A normal operational source term, based on operational reactor experience, as described in American National Standards Institute/American National Standard (ANSI/ANS) N18.1. The source term is also addressed in SRP Section 11.1 for reactor coolant (primary and secondary) and reactor steam design details, and in SRP Section 11.2, “Liquid Waste Management System,” and SRP Section 11.3, “Gaseous Waste Management System,” for system design features used to process and treat liquid and gaseous effluents before being released or recycled. This source term is used to meet the specific regulatory dose and effluent release requirements of 10 CFR 20 and 10 CFR 50 Appendix I. The application also commits to complying with RG 1.206, and NUREG 0800 - SRP 11.1, and utilizing the required ANSI Standard as the source term.

The staff attempted to verify the source term listed in the applicants’ submittal in DCD Table 11.1-9. When comparing the radionuclides in the ANSI 18.1, 1999 Table 6 Numerical Values - Concentrations in Principal Fluid Streams of the Reference PWR with U-Tube Steam Generators (uCi/g) and DCD Table 11.1-9 Expected Specific Activities of Reactor Coolant During Normal Operation, with a footnote (1) denoting ANSI/ANS 18.1, in the applicants submittal there were major differences. Table 11.1-9 references the ANSI Standard 18.1, however the radionuclide values do not agree. Out of the 56 radionuclides, 20 agree within 4% of the ANSI value, 31 are greater than 4%, and 6 are greater than 12.99% of the ANSI value. These values are used along with the operational parameters in DCD Table 11.1-1 under the Normal Operation values, again with a footnote designating ANSI 18.1 as the reference for the expected source term. These table values are used as input into the NUEG-0017, PWR-GALE Code as RCS activity along with assumed operational data to develop the annual curies per year that will be estimated to be released in the liquid and gaseous effluents from this plant design. These effluent values are then carried on to calculate effluent doses to the environment attributed to this reactor design.

For DCD Table 11.1-9 and Section 11.1, please provide the following information:

1. Conversion to Bq/g is not required when submitting information to the NRC per 10 CFR 20.2101. Please revise the DCD Table values to uCi/g in the application.

2. Provide discussion in DCD Section 11.1 as to why the values submitted in DCD Table 11.1-9 do not agree with ANSI/ANS 18.1, 1999, or revise the table and section to reflect the ANSI values.

3. Provide and submit a plan to revise the DCD Table 11.1-9 values, if necessary, all subsequent tables, DCD sections (11.2, 11.3, etc.) that rely on the values listed in DCD Table 11.1-9, and calculations that utilize the Table 11.1-9 values.

Please address these items and provide a mark up for the proposed DCD changes.

Response

1. DCD Table 11.1-9 will be revised to use both Bq/g and μCi/g.

2. According to section 3.2 of ANSI/ANS-18.1, 1999, if any parameter, such as power level, flow rate, or fluid quantity, differs from the values of reference reactor, adjustment factors are needed to modify the radionuclide concentrations in Table 6 of ANSI/ANS-18.1, 1999.

   The expected specific activities of the reactor coolant for APR1400 design, provided in Table 11.1-9, are calculated by multiplying the adjustment factors for APR1400 design and the reference activities of ANSI/ANS-18.1, 1999.

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   The parameters are listed in the following table.

   Values used in determining adjustment factors for APR1400 design are listed in the following table.

Description	Symbol	Element Class					
		1	2	3	4	5	6
Fraction of material removed in passing through the cation demineralizer	NA	0	0	0.9	0	0	0.9
Fraction of material removed in passing through the purification demineralizer	NB	0	0.99	0.5	0	0	0.98

The description of the element classification used in the ANSI/ANS-18.1 is shown as follows.

Class 1 : Noble gases
Class 2 : Halogens
Class 3 : Cesium, Rubidium
Class 4 : Nitrogen
Class 5 : Tritium
Class 6 : Other nuclides

Adjustment factors (f_i) are calculated based on the following equations given in Table 11 of ANSI/ANS-18.1, 1999.

$f_i = [P \cdot WP_n \cdot (R_{ni}+\lambda)]/[WP \cdot P_n \cdot (R_i+\lambda)]$,

where
The subscript 'i' denotes element classes 1, 2, 3, or 6,
The subscript 'n' is denoting the parameters of the reference plant,
f_4 is 1.0 for N-16 (provided in Table 11 of ANSI/ANS-18.1),
f_5 is 1.0 for H-3 (provided in Table 11 of ANSI/ANS-18.1),
λ is the radionuclide decay constant.

Removal rates (R_i) are calculated based on the following equations given in Table 9 of ANSI/ANS-18.1, 1999.

$R_1 = [FB + (FD - FB) \cdot Y] / WP$,
$R_2 = [(FD \cdot NB_2) + (1 - NB_2)(FB + FA \cdot NA_2)] / WP$,
$R_3 = [(FD \cdot NB_3) + (1 - NB_3)(FB + FA \cdot NA_3)] / WP$,
$R_6 = [(FD \cdot NB_6) + (1 - NB_6)(FB + FA \cdot NA_6)] / WP$.

The removal rates are as follows

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(Parameters of the reference plant, denoted by subscript 'n', are obtained from Table 9 of ANSI/ANS-18.1, 1999.)

The adjustment factors for the APR1400 design were calculated and provided in the table below.

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* The Ba-137m activity is assumed to be identical with Cs-137 activity. This nuclide is in secular equilibrium.

Because the values of DCD Table 11.1-9 are calculated by multiplying the adjustment factors and the reference activities of ANSI/ANS-18.1, 1999, it is reasonable that the

values submitted in DCD Table 11.1-9 do not agree with ANSI/ANS 18.1, 1999. DCD Section 11.1.2.1 will be revised to include wording stating that adjustment factors were used to calculate the values provided in Table 11.1-9.

3. Due to the response provided to Question 2, there is no plan to revise the DCD Table 11.1-9 values at this time.

Impact on DCD

DCD Table 11.1-9 will be revised as indicated on the Attachment 1.
DCD section 11.1.2.1 will be revised as indicated on the Attachment 2.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical/Environmental Reports

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

Impact on Technical Specifications

There is no impact on the Technical Specifications.

This table should be replaced with table on page 2 of Attachment 1.

Table 11.1-9

Expected Specific Activities of Reactor Coolant During Normal Operation
(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).

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

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

of 33,085 L (1,168 ft^3). The design basis specific activities of gaseous sources vented from the EDT to the GRS are provided in Table 11.1-8.

11.1.2 Expected Source Term

11.1.2.1 Reactor Coolant Activities

The expected specific activities in the reactor coolant are calculated in accordance with ANSI/ANS 18.1 using an adjustment factor to take into account the normal operating parameters provided in Table 11.1-1.

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 AOOs. ~~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 SG liquid and in the main steam during the normal operation are determined using the method described in Subsection 11.1.1.3. The SG 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.