Tennessee Valley Authority, 1101 Market Street, Chattanooga, TN 37402

CNL-16-191

December 2, 2016

10 CFR 2.101
10 CFR 52.15

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Clinch River Nuclear Site
NRC Project No. 785

Subject: Submittal of Supplemental Information Regarding Radiation Protection and Accident Consequences in Support of Early Site Permit Application for Clinch River Nuclear Site

References:

1. Letter from TVA to NRC, CNL-16-081, "Application for Early Site Permit for Clinch River Nuclear Site," dated May 12, 2016

2. Letter from TVA to NRC, CNL-16-103, "Submittal of Calculation Input and Output Files in Support of Early Site Permit Application for Clinch River Nuclear Site," dated June 23, 2016

3. Letter from TVA to NRC, CNL-16-134, "Schedule for Submittal of Supplemental Information in Support of Early Site Permit Application for Clinch River Nuclear Site," dated August 11, 2016

By letter dated May 12, 2016 (Reference 1), Tennessee Valley Authority (TVA) submitted an application for an early site permit for the Clinch River Nuclear (CRN) Site in Oak Ridge, TN. By letter dated June 23, 2016 (Reference 2), TVA provided input and output files for the GASPAR II and LADTAP II analyses to support the NRC staff's review. In addition, and consistent with interactions with NRC staff, TVA identified certain aspects of the application that it intends to supplement. By letter dated August 11, 2016 (Reference 3), TVA provided a plan for submitting the identified supplemental information.

The enclosure to this letter contains a description of the supplemental information related to radiation protection and accident consequences, including a markup of the affected Site Safety Analysis Report sections. These changes will be incorporated in a future revision of the early site permit application.

There are no new regulatory commitments associated with this submittal. If any additional information is needed, please contact Dan Stout at (423) 751-7642.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 2nd day of December 2016.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosure:

Supplemental Information Regarding Radiation Protection and Accident Consequences

cc (with enclosures):

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ENCLOSURE

Supplemental Information Regarding Radiation Protection and Accident Consequences

By letter dated May 12, 2016 (Reference 1), Tennessee Valley Authority (TVA) submitted an application for an early site permit for the Clinch River Nuclear (CRN) Site in Oak Ridge, TN. By letter dated June 23, 2016 (Reference 2), TVA provided input and output files for the GASPAR II and LADTAP II analyses to support the NRC staff's review. In addition, and consistent with interactions with NRC staff, TVA identified certain aspects of the application that it intends to supplement. By letter dated August 11, 2016 (Reference 3), TVA provided a plan for submitting the identified supplemental information.

This enclosure contains supplemental information related to radiation protection and accident consequences, including a markup of the affected Site Safety Analysis Report (SSAR) sections. These changes will be incorporated in a future revision of the early site permit application (ESPA).

Supplement Item (from Reference 3)

TVA will provide a markup of the applicable ESPA sections to clarify the basis for deriving the annual normal liquid and normal gaseous radioactive effluent releases provided in the plant parameter envelope and the methodology used to compute the daughter products and maximum decay activities for the accidental liquid radwaste release source term.

Supplemental Information

The guidance in Nuclear Energy Institute (NEI) 10-01 was used to derive the plant parameter envelope (PPE) source terms for annual normal liquid and gaseous radioactive effluent release. The guidance in NEI 10-01 was also used to derive the PPE source terms for accidental liquid and gaseous radioactive release.

Attachment 1 to this enclosure provides the following changes to the SSAR:

- Section 1.11 is being revised to indicate that the basis for each plant parameter in Table 2.0-2 is provided in the SSAR section indicated in the table for that plant parameter.
- Subsection 2.4.13.1 is being revised to provide a description of the methodology used to compute maximum decay activities for the accidental liquid radwaste release source term. (Note that SSAR Subsection 2.4.13.5.2 discusses the methodology used to determine the daughter products associated with the initial tank inventory. Therefore, no supplemental information has been provided with respect to the determination of daughter products for the accidental liquid radwaste release source term.)
- Subsection 11.2.3.1 is being revised to specify that the guidance in NEI 10-01 was used to develop the surrogate plant source term for the liquid radioactive effluent release.
- Subsection 11.3.3.1 is being revised to specify that the guidance in NEI 10-01 was used to develop the surrogate plant source term for the gaseous radioactive effluent release.

Additional detail regarding the basis for the source terms for annual normal liquid and gaseous radioactive effluent release, as well as the basis for the source terms for accidental liquid and gaseous radioactive release, is provided in Attachments 2, 3, 4, and 5 to this enclosure.

ENCLOSURE

Supplemental Information Regarding Radiation Protection and Accident Consequences

References:

1. Letter from TVA to NRC, CNL-16-081, "Application for Early Site Permit for Clinch River Nuclear Site," dated May 12, 2016

2. Letter from TVA to NRC, CNL-16-103, "Submittal of Calculation Input and Output Files in Support of Early Site Permit Application for Clinch River Nuclear Site," dated June 23, 2016

3. Letter from TVA to NRC, CNL-16-134, "Schedule for Submittal of Supplemental Information in Support of Early Site Permit Application for Clinch River Nuclear Site," dated August 11, 2016

Attachments:

1. Site Safety Analysis Report Markups
2. Basis of Source Term for Normal Liquid Radioactive Effluent Release
3. Basis of Source Term for Normal Gaseous Radioactive Effluent Release
4. Basis of Source Term for Accidental Liquid Radioactive Effluent Release
5. Basis of Source Term for Accidental Gaseous Radioactive Effluent Release

SSAR Section 1.11 is being revised as indicated. Strikethroughs indicate text to be deleted. Underlines indicate text to be added.

1.11 OVERVIEW OF REACTOR TYPES

Four conceptual, light-water cooled, small modular reactor (SMR) designs were used to create a "surrogate plant" as defined in NEI 10-01, Industry Guideline for Developing a Plant Parameter Envelope in Support of an Early Site Permit (Reference 1.11-1) and to develop the site-related design parameter values listed in Table 2.0-2 of Chapter 2. The basis for each plant parameter is provided in the SSAR section indicated in Table 2.0-2 for that plant parameter. The~~se~~ reactor designs are:

- BWXT mPower™ (Generation mPower LLC design)
- NuScale (NuScale Power, LLC, design)
- SMR-160 (Holtec SMR, LLC, design)
- Westinghouse SMR (Westinghouse Electric Company, LLC, design)

All four designs are described as passively safe with minimal or no reliance on offsite power, offsite water, or operator action for safety. Based on design features, these designs eliminate various conventional design basis events (e.g., large-break LOCAs precluded by elimination of large bore piping). All but the SMR-160 (Holtec SMR, LLC, design) are integral pressurized water reactors (iPWRs); that is, pressurized water reactor (PWR) designs in which the primary coolant system and all (or most) of its components (i.e., pressurizer, steam generators, and reactor coolant pumps, where applicable) are enclosed in one pressure vessel.

Attachment 1
Site Safety Analysis Report Markups

SSAR Subsection 2.4.13.1 is being revised as indicated. Underlines indicate text to be added.

2.4.13.1 Accident Source

It is postulated that a liquid radwaste tank outside of containment or outdoors ruptures with its contents released to the environment. The maximum tank volume with liquid radioactive waste is 10,000 gallons. The <u>initial</u> postulated radionuclide inventory of the spill is shown on Table 2.0-5. A simplified release scenario is assumed, in which 80 percent of the tank volume is transferred instantaneously to the groundwater at a point within the power block area, and no credit is taken for the time that radionuclides may take (and the associated radioactive decay) to travel from the Liquid Waste Management System (LWMS) tank to the saturated zone. Analyzing a release of 80 percent of the tank volume is based on the guidance in DC/COL-ISG-013 (Reference 2.4.13-1) which states, "The radionuclide inventory for the tank and its components assumed to fail should be based on a conservative estimate of 80 percent capacity of that tank and its components." It is anticipated that a postulated radionuclide release will be mixed with groundwater of the CRN Site and will travel toward the Clinch River arm of the Watts Bar Reservoir.

<u>Each of the four reactor vendors provided source term information for the accidental release of radioactive liquids, consistent with the guidance in NEI 10-01 (Reference 2.4.13-19). The source terms of the four SMR vendors were evaluated. The source term concentration for one vendor was found to be more conservative. On this basis, the source term associated with this vendor was adopted as the surrogate plant values. However, based upon the large amount of conservatism included in the surrogate plant values as compared to the other SMR designs, a lower activity was used for Zr-95 and Sb-95. In addition, the surrogate plant values were compared to the PSEG ESPA source term values (Reference 2.4.13-20) to assess the reasonableness of the surrogate plant source terms. This comparison concluded that the surrogate plant source terms are conservative when compared to source terms for large light water reactors and are considered to be reasonable for use.</u>

DC/COL-ISG-013 (Reference 2.4.13-1) also indicates that an applicant may take credit for mitigating design features provided they demonstrate that "such features are durable and passive and that the receiving system has the storage capacity to hold the expected volume of liquid wastes." As a reactor technology has not been selected, no mitigation design features are considered in this analysis. Mitigation features included in the design of the reactor technology selected are addressed in the Combined License Application.

As a result of adding text to SSAR Subsection 2.4.13.1, the following references are being added to SSAR Subsection 2.4.13.8:

2.4.13.8 References

<u>2.4.13-19. NEI 10-01, "Industry Guidance for Developing a Plant Parameter Envelope in Support of an Early Site Permit," Revision 1, May 2012.</u>

<u>2.4.13-20. PSEG Power, LLC, "Application for Early Site Permit for the PSEG Site," Revision 4, June 5, 2015.</u>

Attachment 1
Site Safety Analysis Report Markups

SSAR Subsection 11.2.3.1 is being revised as indicated. Underlines indicate text to be added.

11.2.3.1 Exposure Pathways

Small quantities of radioactive liquids would be discharged to the Clinch River arm of the Watts Bar Reservoir, from the discharge structure at Clinch River Mile 15.5 shown in Figure 1.2-2, during normal operation of the nuclear units at the CRN Site. The impact of these releases on individuals and the general population in the vicinity of the site is evaluated by considering the major exposure pathways from the release point to the receptors of interest. The major exposure pathways are those that could yield the highest radiological doses for a given receptor. The relative importance of an exposure pathway is based on the type and amount of radioactivity released, the environmental transport mechanism, and the consumption or usage factors at the receptor.

The source term for the liquid radioactive waste effluents from normal plant operations at the CRN Site is provided in Table 2.0-6. As compliance is evaluated on a per-unit basis in some cases (i.e., 10 CFR 50, Appendix I) and on a per-site basis in others (i.e., 10 CFR 20, Appendix B, and 40 CFR 190), a determination of the source term with releases from one unit and for the site (all units) was required. In both cases, the source term for the surrogate plant is a composite of the four small modular reactor (SMR) technologies being considered. The source term associated with the release from one unit is a composite developed by selecting the highest activity for each of the radionuclides from the information for the four SMR technologies, consistent with the guidance provided in NEI 10-01 (Reference 11.2-4). The source term associated with the site was determined by selecting the highest per-radionuclide site activity of those calculated for each of the SMR technologies (determined by multiplying the per-radionuclide activity for a specific SMR unit by the maximum number of units being considered for that SMR technology). In some cases, based upon the maturity of the source term and the amount of conservatism, a lower activity for certain radionuclides was used. An evaluation was performed to ensure that the adjusted source terms utilized to evaluate the dose consequences are conservative compared to source terms for large light water reactors (scaled to a comparable power output) and are considered to be reasonable for use.

As a result of adding text to SSAR Subsection 11.2.3.1, the following reference is being added to SSAR Subsection 11.2.4:

11.2.4 References

11.2-4. NEI 10-01, "Industry Guidance for Developing a Plant Parameter Envelope in Support of and Early Site Permit," Revision1, May 2012.

SSAR Subsection 11.3.3.1 is being revised as indicated. Underlines indicate text to be added.

11.3.3.1 Exposure Pathways

Small quantities of radioactive gases would be discharged to the environment during normal operation of nuclear units at the CRN Site. The impact of these releases on individuals and the general population in the vicinity of the site is evaluated by considering the most important pathways from the release to the receptors of interest. The major pathways are those that could yield the highest radiological doses for a given receptor. The relative importance of a pathway is based on the type and amount of radioactivity released, the environmental transport mechanism, and the consumption or usage factors at the receptor.

The source term for the gaseous radioactive waste effluents from normal plant operations at the CRN Site is provided in Table 2.0-4. As compliance is evaluated on a per-unit basis in some cases (i.e., 10 CFR 50, Appendix I) and on a per-site basis in others (i.e., 10 CFR 20, Appendix B, and 40 CFR 190), a determination of the source term with releases from one unit and for the site (all units) was required. In both cases, the source term <u>for the surrogate plant</u> is a composite of the four small modular reactor (SMR) technologies being considered. The source term associated with the release from one unit is a composite developed by selecting the highest activity for each of the radionuclides from the information for the four SMR technologies<u>, consistent with the guidance provided in NEI 10-01 (Reference 11.3-2)</u>. The source term associated with the site was determined by selecting the highest per-radionuclide site activity of those calculated for each of the SMR technologies (determined by multiplying the per-radionuclide activity for a specific SMR unit by the maximum number of units being considered for that SMR technology). In some cases, the maturity of the source term resulted in an excessive amount of conservatism. In these instances, a lower activity for certain radionuclides was used. An evaluation was performed to ensure that the source terms utilized to evaluate the dose consequences are conservative compared to source terms for large light water reactors (scaled to a comparable power output) and are considered to be reasonable for use.

As a result of adding text to SSAR Subsection 11.3.3.1, the following reference is being added to SSAR Subsection 11.3.4:

11.3.4 References

<u>11.3-2. NEI 10-01, "Industry Guidance for Developing a Plant Parameter Envelope in Support of and Early Site Permit," Revision1, May 2012.</u>

Attachment 2
Basis of Source Term for Normal Liquid Radioactive Effluent Release

Guidance from Nuclear Energy Institute (NEI) 10-01 was used to determine the normal liquid release source terms. The definition of the parameter noted in NEI 10-01 is "The annual activity, by radionuclide, contained in routine plant liquid effluent streams." The release activities are used in the Environmental Report (Section 3.5, Liquid Waste Management System) and the Site Safety Analysis Report (Subsection 11.2.3, Liquid Radioactive Releases) to describe the liquid radioactive waste management subsystem.

In accordance with the guidelines of NEI 10-01, source term information was obtained for the normal liquid effluent release activities. The small modular reactor (SMR) vendors provided the information with various levels of computational maturity. In some instances, values of release activity were overly conservative and adjustments were made on a case by case basis.

Tennessee Valley Authority created a composite table on a per unit basis and on a total site basis. The site basis multiplied the activities of each vendor by the anticipated number of units for the Clinch River Nuclear (CRN) Site (e.g. mPower x 4, NuScale x 12, etc). Additionally, for conservatism, one vendors activities were increased by 10 percent (%) to compute the composite unit and site tables.

The composite tables of source terms for the per unit basis and total site bases were then utilized to compute the offsite dose consequences. Table 10-A was developed as a composite table that includes annual release activity, by radionuclide, on a per unit basis. Table 10-B was developed as a composite table that includes annual release activity, by radionuclide, on a per site basis. The results of the computations indicate that all regulatory acceptance criteria are met.

During engineering analysis of this parameter, it was identified that some of the radionuclide annual release activities contained in one vendor's values included excessive conservatism. Adjustments to some of the annual release activity values for that vendor were made that caused the annual release activity values in the composite tables to no longer bound the values provided by that vendor. On a single unit basis, carbon-14 (C-4) and cobalt-58 (Co-58) were adjusted to reduce conservatism. On a site basis, tritium (H-3), C-14, cobalt-58 and cobalt-60 (Co-60) were adjusted to reduce conservatism. Additionally, in the development of the composite table, iron-55 (Fe-55) and nickel-63 (Ni-63) were rounded down by one ten-thousandth and one thousandth, respectively. The specific annual activities for the radionuclides that are not bounded are highlighted in red in Tables 10-A and 10-B. These adjustments are acceptable considering the limited development of the fuel and waste management design for the vendor at the time the CRN dose analysis was performed. Subsequent to completion of the CRN analysis, this vendor performed a review of the composite table values used in the plant parameter envelop (PPE) and confirmed that the values used in the composite tables bound the most recent values predicted. This change resulted from further advancements in their fuel and waste management design.

Evaluation of Source Terms for Reasonableness

For comparison of the adjusted values, normal liquid source terms from the PSEG Early Site Permit Application (ESPA) were scaled by reactor thermal power to the reactor thermal power considered in the CRN ESPA. The adjusted annual release activity values in the CRN composite table and the scaled PSEG annual release activity values were then compared. The scaled PSEG values are representative of the source terms for the technologies considered in the CRN ESPA and provide an adequate means for comparison. (See Table 1 and Table 2.)

Comparison to the PSEG values is appropriate because SMR waste management system designs are not expected to significantly change from existing industry designs.

The PSEG ESPA considers four large light water reactor technologies, consisting of a General Electric Nuclear Energy Advanced Boiling Water Reactor, Westinghouse Electric Company Advanced Passive 1000 (AP1000), AREVA NP, Inc. U.S. EPR, and the Mitsubishi Heavy Industries, Ltd. U.S. Advanced Pressurized-Water Reactor (US-APWR). The methodology used in the PSEG ESPA to determine normal liquid source terms is a composite table developed from the four reactor technologies considered. This composite table methodology is considered conservative as no single vendor contains the highest source terms for all radionuclides.

The equation used for scaling the unit and site values is shown below. For the unit basis, the CRN thermal power was considered to be 800 MWth and the PSEG thermal power was considered to be 3,400 MWth. For the site basis, the CRN thermal power was considered 2,400 MWth and the PSEG thermal power was considered to be 6,800 MWth, assuming two units for the PSEG site. The thermal powers considered for PSEG are conservative due to the inclusion of a U.S. EPR reactor, (4,500 MWth) in the PSEG composite table of normal liquid effluent release source terms.

$$Annual\ Activity_{Scaled} = Annual\ Activity_{PSEG}\frac{CRN\ Thermal\ Power}{PSEG\ Thermal\ Power}$$

The scaled values determined from the PSEG ESPA values for the unit and site basis are shown in Tables 1 and 2 below. Comparison of the scaled PSEG values to the values in the CRN composite table (Table 1) shows that all of the adjusted annual activities in the CRN composite table, listed above, are more conservative (i.e., larger) than the scaled PSEG values, for a unit basis.

For the site basis, Table 2 shows that the surrogate values for carbon-14 and cobalt-58 are larger than the scaled PSEG values. H-3 and Co-60 the are less than the scaled PSEG values.

However, H-3 considered in the CRN composite table is greater than that determined using the NUREG-0017 methodology. As mentioned in a topical report submitted by one of the vendors, "Effluent Release (GALE Replacement) Methodology," H-3 effluent release rates are determined to be 0.4 Ci/year/MWth with up to 90% of the H-3 released as liquid. For a total site thermal reactor power, assumed 2400 MWth for a bounding PPE methodology, the maximum expected release rate of H-3 in liquid effluents is 864 Ci/year (0.4 Ci/year X 2400MWth X 0.90) based on the NUREG-0017 methodology. The H-3 source term identified in the PPE is 885 Ci/year, conservatively higher than that developed using the NUREG-0017 methodology by approximately 2.4% ([(885-864)/864] X 100%).

The CRN site surrogate value for Co-60 is slightly lower than the scaled PSEG value. The difference in one isotope is within the range of comparison using this methodology and is bounded by the conservative nature of the composite table. As stated earlier, later evaluations performed by the vendor concluded that the values used by the CRN surrogate plant (PPE values) are bounding to their most recent predicted activity release values.

During comparison of the PSEG scaled values to the CRN composite table values, it was noted that PSEG did not evaluate C-14 for normal liquid radioactive releases. C-14 is primarily

released in gaseous effluents, but is conservatively considered in the PPE composite methodology for liquid effluents. The exclusion of C-14 is a commonly accepted practice for the development of liquid radioactive source terms, as shown by previous ESPAs. Additionally, Electric Power Research Institute (EPRI) issued a technical report in 2010, "Estimation of Carbon-14 in Nuclear Power Plant Gaseous Effluents," providing information regarding gaseous and liquid C-14 releases that supports the conclusion that dose effects of C-14 in liquid effluents is negligible.

Basis for Surrogate Plant Representing Vendor Provided Information

In conclusion, the normal liquid radioactive release composite tables are reasonable for the CRN ESPA. Adjusted source term values, other than H-3, are shown to be not unreasonable by the conservative comparison of scaled PSEG values. The values for H-3 in the CRN ESPA composite are shown to not be unreasonable based on a methodology, outlined in NUREG-0017 and a topical report, used to determine H-3 source terms releases. The adjustments to C-14 are shown to be not unreasonable by the conservative inclusion of a C-14 source terms, in light of regulatory guidance and an EPRI technical report statements supporting the exclusion of C-14 from liquid radioactive releases.

Table 1 One Unit (Ci/year)		
Radionuclide	**CRN Release**	**Scaled PSEG**
C-14	8.19E-04	0.00E+00
Co-58	5.20E-03	2.31E-03

Table 2 Site (Ci/year)		
Radionuclide	**CRN Release**	**Scaled PSEG**
H-3	8.85E+02	1.17E+03
C-14	9.83E-03	0.00E+00
Co-58	5.51E-02	6.92E-03
Co-60	8.21E-03	1.09E-02

Attachment 2
Basis of Source Term for Normal Liquid Radioactive Effluent Release

Vendor 1

Table 10 - A
One Unit
Average Annual Normal Liquid Radioactive Release

Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
I-129	3.30E-12	Rh-103m	2.42E-07
I-130	4.62E-06	Rh-105	1.07E-07
I-131	9.13E-04	Rh-106	9.35E-08
I-132	1.43E-04	Ag-110m	5.83E-07
I-133	8.47E-04	Ag-110	8.69E-09
I-134	7.48E-06	Sb-125	1.98E-09
I-135	2.75E-04	Sb-127	1.10E-08
H-3	2.09E+02	Sb-129	4.40E-09
Na-24	3.08E-11	Te-127m	1.43E-06
Cr-51	8.58E-09	Te-127	3.19E-06
Mn-54	8.14E-10	Te-129m	4.84E-06
Fe-55	3.30E-09	Te-129	3.19E-06
Fe-59	2.09E-10	Te-131m	1.03E-05
Co-58	1.32E-09	Te-131	2.31E-06
Co-60	2.86E-10	Te-132	1.08E-04
Zn-65	9.24E-17	Te-134	2.64E-07
Br-82	1.87E-06	Cs-134	1.98E-03
Br-83	3.52E-06	Cs-136	6.16E-04
Br-84	2.31E-07	Cs-137	1.65E-03
Br-85	2.42E-09	Cs-138	3.19E-05
Rb-86	1.87E-05	Ba-137m	5.17E-04
Rb-88	4.40E-05	Ba-139	1.54E-08
Rb-89	2.31E-06	Ba-140	1.65E-06
Sr-89	1.43E-06	La-140	7.81E-07
Sr-90	1.32E-07	La-141	2.20E-08
Sr-91	2.86E-07	La-142	2.97E-09
Sr-92	5.61E-08	Ce-141	2.53E-07
Y-90	4.95E-08	Ce-143	1.43E-07
Y-91m	1.76E-07	Ce-144	2.20E-07
Y-91	1.98E-07	Pr-143	2.31E-07
Y-92	1.32E-07	Pr-144	2.20E-07
Y-93	6.16E-08	Nd-147	9.57E-08
Zr-95	2.75E-07	Np-239	2.09E-06
Zr-97	1.10E-07	Pu-238	6.60E-10
Nb-95	2.75E-07	Pu-239	8.47E-11
Mo-99	2.64E-04	Pu-240	1.07E-10
Tc-99m	2.53E-04	Pu-241	3.19E-08
Tc-99	4.40E-09	Am-241	4.62E-11
Ru-103	2.42E-07	Cm-242	9.46E-09
Ru-105	1.76E-08	Cm-244	4.40E-10
Ru-106	9.46E-08	Total	2.09E+02

Vendor 2

Table 10 - A
One Unit
Average Annual Normal Liquid Radioactive Release

Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
I-129	NA	Sr-92	5.91E-05
I-131	4.53E-03	Y-92	2.25E-04
I-132	2.57E-04	Y-93	1.81E-04
I-133	4.97E-03	Zr-95	1.73E-04
I-134	1.08E-04	Nb-95	2.67E-04
I-135	2.00E-03	Mo-99	3.48E-04
H-3	2.21E+02	Tc-99m	7.57E-04
C-14	NA	Tc-99	NA
Na-24	8.13E-04	Ru-103	6.57E-04
P-32	7.57E-05	Rh-103m	
Cr-51	2.27E-03	Ru-106	9.80E-03
Mn-54	6.00E-04	Rh-106	NA
Mn-56	2.72E-04	Ag-110m	2.40E-04
Co-56	NA	Sb-124	5.73E-05
Co-57	NA	Te-129m	1.60E-05
Co-58	1.31E-03	Te-131m	4.13E-05
Co-60	2.05E-03	Te-132	6.40E-05
Fe-55	1.26E-03	Cs-134	1.60E-03
Fe-59	3.07E-04	Cs-136	2.93E-03
Ni-63	2.27E-04	Cs-137	2.40E-03
Cu-64	1.68E-03	Cs-138	1.07E-07
Zn-65	5.88E-05	Ba-140	7.73E-04
Br-84	2.67E-06	La-140	1.07E-03
Rb-88	3.73E-03	Ce-141	3.96E-05
Rb-89	NA	Ce-143	8.13E-05
Sr-89	4.19E-05	Ce-144	7.47E-04
Sr-90	3.57E-06	Pr-143	1.73E-05
Y-90	NA	W-187	6.13E-05
Sr-91	1.67E-04	Np-239	1.27E-03
Y-91	3.13E-05	Nd-147	2.67E-07
		Pr-144	4.21E-04
		Te-129	4.13E-05
		Te-131	1.01E-05
		Y-91m	6.67E-06

Vendor 3

Table 10 - A
One Unit
Average Annual Normal Liquid Radioactive Release

	Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
	I-129	4.20E-10	Sr-92	3.87E-06
	I-131	1.38E-02	Y-92	9.56E-07
	I-132	6.04E-03	Y-93	1.70E-06
	I-133	2.30E-02	Zr-95	1.83E-04
	I-134	3.26E-03	Nb-95	3.17E-06
	I-135	1.37E-02	Mo-99	3.77E-03
	H-3	1.61E+02	Tc-99m	1.89E-03
**	C-14	6.81E+00	Tc-99	
	Na-24		Ru-103	2.59E-06
	P-32		Rh-103m	3.64E-07
	Cr-51	1.07E-02	Ru-106	7.99E-07
	Mn-54	5.44E-03	Rh-106	1.16E-09
	Mn-56		Ag-110m	2.22E-03
	Co-56		Sb-124	
	Co-57		Te-129m	1.30E-07
**	Co-58	7.39E-03	Te-131m	1.18E-04
	Co-60	7.70E-04	Te-132	1.45E-03
*	Fe-55	4.07E-03	Cs-134	2.87E-03
	Fe-59	9.92E-04	Cs-136	9.00E-04
*	Ni-63	1.54E-02	Cs-137	3.53E-03
	Cu-64		Cs-138	1.18E-03
	Zn-65	1.76E-03	Ba-140	1.96E-05
	Br-84	8.38E-05	La-140	2.83E-06
	Rb-88	4.95E-04	Ce-141	3.04E-06
	Rb-89	5.15E-05	Ce-143	2.39E-06
	Sr-89	1.19E-05	Ce-144	2.42E-06
	Sr-90	1.06E-06	Pr-143	2.76E-06
	Y-90	1.55E-07	W-187	
	Sr-91	8.64E-06	Np-239	2.49E-03
	Y-91	2.44E-06		

Vendor 4

Table 10 - A
One Unit
Average Annual Normal Liquid Radioactive Release

Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
I-129		Sr-92	
I-131		Y-92	
I-132	4.40E-02	Y-93	
I-133	1.70E-04	Zr-95	
I-134	9.90E-04	Nb-95	3.00E-05
I-135	1.00E-05	Mo-99	8.80E-04
H-3		Tc-99m	7.50E-04
C-14		Tc-99	
Na-24	2.80E-03	Ru-103	
P-32		Rh-103m	
Cr-51	2.80E-03	Ru-106	
Mn-54	2.00E-03	Rh-106	
Mn-56		Ag-110m	
Co-56		Sb-124	
Co-57		Te-129m	2.30E-02
Co-58	5.20E-03	Te-131m	6.60E-04
Co-60	6.80E-04	Te-132	4.40E-02
Fe-55	1.50E-03	Cs-134	
Fe-59	3.10E-04	Cs-136	
Ni-63		Cs-137	
Cu-64		Cs-138	
Zn-65	6.30E-04	Ba-140	1.60E-02
Br-84		La-140	
Rb-88	1.20E-04	Ce-141	
Rb-89		Ce-143	
Sr-89		Ce-144	
Sr-90		Pr-143	
Y-90		W-187	2.10E-04
Sr-91	1.00E-05	Np-239	3.70E-04
Y-91			

Composite Table

Table 10 - A
One Unit
Average Annual Normal Liquid Radioactive Release

Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
I-129	4.20E-10	Ru-105	1.76E-08
I-130	4.62E-06	Rh-103m	3.64E-07
I-131	1.38E-02	Ru-106	9.80E-03
I-132	4.40E-02	Rh-106	9.35E-08
I-133	2.30E-02	Rh-105	1.07E-07
I-134	3.26E-03	Ag-110m	2.22E-03
I-135	1.37E-02	Ag-110	8.69E-09
H-3	2.21E+02	Sb-124	5.73E-05
C-14	8.19E-04	Sb-125	1.98E-09
Na-24	2.80E-03	Sb-127	1.10E-08
P-32	7.57E-05	Sb-129	4.40E-09
Cr-51	1.07E-02	Te-127m	1.43E-06
Mn-54	5.44E-03	Te-127	3.19E-06
Mn-56	2.72E-04	Te-129m	2.30E-02
Co-58	5.20E-03	Te-129	3.19E-06
Co-60	2.05E-03	Te-131m	6.60E-04
Fe-55	4.06E-03	Te-131	2.31E-06
Fe-59	9.92E-04	Te-132	4.40E-02
Ni-63	1.53E-02	Te-134	2.64E-07
Cu-64	1.68E-03	Cs-134	2.87E-03
Zn-65	1.76E-03	Cs-136	2.93E-03
Br-82	1.87E-06	Cs-137	3.53E-03
Br-83	3.52E-06	Cs-138	1.18E-03
Br-84	8.38E-05	Ba-137m	5.17E-04
Br-85	2.42E-09	Ba-139	1.54E-08
Rb-86	1.87E-05	Ba-140	1.60E-02
Rb-88	3.73E-03	La-140	1.07E-03
Rb-89	5.15E-05	La-141	2.20E-08
Sr-89	4.19E-05	La-142	2.97E-09
Sr-90	3.57E-06	Ce-141	3.96E-05
Sr-91	1.67E-04	Ce-143	8.13E-05
Sr-92	5.91E-05	Ce-144	7.47E-04
Y-90	1.55E-07	Pr-143	1.73E-05
Y-91	3.13E-05	Pr-144	4.21E-04
Y-91m	6.67E-06	Nd-147	2.67E-07
Y-92	2.25E-04	Np-239	2.49E-03
Y-93	1.81E-04	Pu-238	6.60E-10
Zr-95	1.83E-04	Pu-239	8.47E-11
Zr-97	1.10E-07	Pu-240	1.07E-10
Nb-95	2.67E-04	Pu-241	3.19E-08
Mo-99	3.77E-03	Am-241	4.62E-11
Tc-99m	1.89E-03	Cm-242	9.46E-09
Tc-99	4.40E-09	Cm-244	4.40E-10
Ru-103	6.57E-04	W-187	2.10E-04
		Total	2.22E+02

Notes:
Items highlighted in Yellow are bounding values for surrogate plant (Composite Table)
Items highlighted in Red are not bounded by surrogate plant value (Composite Table)

* Value inadvertently rounded down
** Value contains excessive conservatism

Attachment 2
Basis of Source Term for Normal Liquid Radioactive Effluent Release

Vendor 1

Table 10 - B Site Average Annual Normal Liquid Radioactive Release			
Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
I-129	1.32E-11	Rh-103m	9.68E-07
I-130	1.85E-05	Rh-105	4.27E-07
I-131	3.65E-03	Rh-106	3.74E-07
I-132	5.72E-04	Ag-110m	2.33E-06
I-133	3.39E-03	Ag-110	3.48E-08
I-134	2.99E-05	Sb-125	7.92E-09
I-135	1.10E-03	Sb-127	4.40E-08
H-3	8.36E+02	Sb-129	1.76E-08
Na-24	1.23E-10	Te-127m	5.72E-06
Cr-51	3.43E-08	Te-127	1.28E-05
Mn-54	3.26E-09	Te-129m	1.94E-05
Fe-55	1.32E-08	Te-129	1.28E-05
Fe-59	8.36E-10	Te-131m	4.14E-05
Co-58	5.28E-09	Te-131	9.24E-06
Co-60	1.14E-09	Te-132	4.31E-04
Zn-65	3.70E-16	Te-134	1.06E-06
Br-82	7.48E-06	Cs-134	7.92E-03
Br-83	1.41E-05	Cs-136	2.46E-03
Br-84	9.24E-07	Cs-137	6.60E-03
Br-85	9.68E-09	Cs-138	1.28E-04
Rb-86	7.48E-05	Ba-137m	2.07E-03
Rb-88	1.76E-04	Ba-139	6.16E-08
Rb-89	9.24E-06	Ba-140	6.60E-06
Sr-89	5.72E-06	La-140	3.12E-06
Sr-90	5.28E-07	La-141	8.80E-08
Sr-91	1.14E-06	La-142	1.19E-08
Sr-92	2.24E-07	Ce-141	1.01E-06
Y-90	1.98E-07	Ce-143	5.72E-07
Y-91m	7.04E-07	Ce-144	8.80E-07
Y-91	7.92E-07	Pr-143	9.24E-07
Y-92	5.28E-07	Pr-144	8.80E-07
Y-93	2.46E-07	Nd-147	3.83E-07
Zr-95	1.10E-06	Np-239	8.36E-06
Zr-97	4.40E-07	Pu-238	2.64E-09
Nb-95	1.10E-06	Pu-239	3.39E-10
Mo-99	1.06E-03	Pu-240	4.27E-10
Tc-99m	1.01E-03	Pu-241	1.28E-07
Tc-99	1.76E-08	Am-241	1.85E-10
Ru-103	9.68E-07	Cm-242	3.78E-08
Ru-105	7.04E-08	Cm-244	1.76E-09
Ru-106	3.78E-07	Total	8.36E+02

Notes:
Items highlighted in Yellow are bounding values for surrogate plant (Composite Table)
Items highlighted in Red are not bounded by surrogate plant value (Composite Table)

* Value inadvertently rounded down
** Value contains excessive conservatism

Vendor 2

Table 10 - B Site Average Annual Normal Liquid Radioactive Release			
Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
I-129	NA	Sr-92	2.36E-04
I-131	1.81E-02	Y-92	9.01E-04
I-132	1.03E-03	Y-93	7.25E-04
I-133	1.99E-02	Zr-95	6.93E-04
I-134	4.32E-04	Nb-95	1.07E-03
I-135	8.00E-03	Mo-99	1.39E-03
H-3	8.85E+02	Tc-99m	3.03E-03
C-14	NA	Tc-99	NA
Na-24	3.25E-03	Ru-103	2.63E-03
P-32	3.03E-04	Rh-103m	
Cr-51	9.07E-03	Ru-106	3.92E-02
Mn-54	2.40E-03	Rh-106	NA
Mn-56	1.09E-03	Ag-110m	9.60E-04
Co-56	NA	Sb-124	2.29E-04
Co-57	NA	Te-129m	6.40E-05
Co-58	5.23E-03	Te-131m	1.65E-04
Co-60	8.21E-03	Te-132	2.56E-04
Fe-55	5.05E-03	Cs-134	6.40E-03
Fe-59	1.23E-03	Cs-136	1.17E-02
Ni-63	9.07E-04	Cs-137	9.60E-03
Cu-64	6.72E-03	Cs-138	4.27E-07
Zn-65	2.35E-04	Ba-140	3.09E-03
Br-84	1.07E-05	La-140	4.27E-03
Rb-88	1.49E-02	Ce-141	1.58E-04
Rb-89	NA	Ce-143	3.25E-04
Sr-89	1.67E-04	Ce-144	2.99E-03
Sr-90	1.43E-05	Pr-143	6.93E-05
Y-90	NA	W-187	2.45E-04
Sr-91	6.67E-04	Np-239	5.06E-03
Y-91	1.25E-04	Nd-147	1.07E-06
		Pr-144	1.69E-03
		Te-129	1.65E-04
		Te-131	4.05E-05
		Y-91m	2.67E-05

Vendor 3

	Table 10 - B Site Average Annual Normal Liquid Radioactive Release			
	Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
	I-129	5.04E-09	Sr-92	4.64E-05
	I-131	1.66E-01	Y-92	1.15E-05
	I-132	7.25E-02	Y-93	2.04E-05
	I-133	2.76E-01	Zr-95	2.20E-03
	I-134	3.91E-02	Nb-95	3.80E-05
	I-135	1.64E-01	Mo-99	4.52E-02
**	H-3	1.93E+03	Tc-99m	2.27E-02
**	C-14	8.17E+01	Tc-99	
	Na-24		Ru-103	3.11E-05
	P-32		Rh-103m	4.37E-06
	Cr-51	1.28E-01	Ru-106	9.59E-06
	Mn-54	6.53E-02	Rh-106	1.39E-08
	Mn-56		Ag-110m	2.66E-02
	Co-56		Sb-124	
	Co-57		Te-129m	1.56E-06
**	Co-58	8.87E-02	Te-131m	1.42E-03
**	Co-60	9.24E-03	Te-132	1.74E-02
*	Fe-55	4.88E-02	Cs-134	3.44E-02
	Fe-59	1.19E-02	Cs-136	1.08E-02
*	Ni-63	1.85E-01	Cs-137	4.24E-02
	Cu-64		Cs-138	1.42E-02
	Zn-65	2.11E-02	Ba-140	2.35E-04
	Br-84	1.01E-03	La-140	3.40E-05
	Rb-88	5.94E-03	Ce-141	3.65E-05
	Rb-89	6.18E-04	Ce-143	2.87E-05
	Sr-89	1.43E-04	Ce-144	2.90E-05
	Sr-90	1.27E-05	Pr-143	3.31E-05
	Y-90	1.86E-06	W-187	
	Sr-91	1.04E-04	Np-239	2.99E-02
	Y-91	2.93E-05		

Vendor 4

Table 10 - B Site Average Annual Normal Liquid Radioactive Release			
Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
I-129		Sr-92	
I-131		Y-92	
I-132	1.32E-01	Y-93	
I-133	5.10E-04	Zr-95	
I-134	2.97E-03	Nb-95	9.00E-05
I-135	3.00E-05	Mo-99	2.64E-03
H-3		Tc-99m	2.25E-03
C-14		Tc-99	
Na-24	8.40E-03	Ru-103	
P-32		Rh-103m	
Cr-51	8.40E-03	Ru-106	
Mn-54	6.00E-03	Rh-106	
Mn-56		Ag-110m	
Co-56		Sb-124	
Co-57		Te-129m	6.90E-02
Co-58	1.56E-02	Te-131m	1.98E-03
Co-60	2.04E-03	Te-132	1.32E-01
Fe-55	4.50E-03	Cs-134	
Fe-59	9.30E-04	Cs-136	
Ni-63		Cs-137	
Cu-64		Cs-138	
Zn-65	1.89E-03	Ba-140	4.80E-02
Br-84		La-140	
Rb-88	3.60E-04	Ce-141	
Rb-89		Ce-143	
Sr-89		Ce-144	
Sr-90		Pr-143	
Y-90		W-187	6.30E-04
Sr-91	3.00E-05	Np-239	1.11E-03
Y-91			

Composite Table

Table 10 - B Site Average Annual Normal Liquid Radioactive Release			
Radionuclide	Release Ci/yr	Radionuclide	Release Ci/yr
I-129	5.04E-09	Ru-105	7.04E-08
I-130	1.85E-05	Rh-103m	4.37E-06
I-131	1.66E-01	Ru-106	3.92E-02
I-132	1.32E-01	Rh-106	3.74E-07
I-133	2.76E-01	Rh-105	4.27E-07
I-134	3.91E-02	Ag-110m	2.66E-02
I-135	1.64E-01	Ag-110	3.48E-08
H-3	8.85E+02	Sb-124	2.29E-04
C-14	9.83E-03	Sb-125	7.92E-09
Na-24	8.40E-03	Sb-127	4.40E-08
P-32	3.03E-04	Sb-129	1.76E-08
Cr-51	1.28E-01	Te-127m	5.72E-06
Mn-54	6.53E-02	Te-127	1.28E-05
Mn-56	1.09E-03	Te-129m	6.90E-02
Co-58	5.51E-02	Te-129	1.28E-05
Co-60	8.21E-03	Te-131m	1.98E-03
Fe-55	4.87E-02	Te-131	9.24E-06
Fe-59	1.19E-02	Te-132	1.32E-01
Ni-63	1.84E-01	Te-134	1.06E-06
Cu-64	6.72E-03	Cs-134	3.44E-02
Zn-65	2.11E-02	Cs-136	1.17E-02
Br-82	7.48E-06	Cs-137	4.24E-02
Br-83	1.41E-05	Cs-138	1.42E-02
Br-84	1.01E-03	Ba-137m	2.07E-03
Br-85	9.68E-09	Ba-139	6.16E-08
Rb-86	7.48E-05	Ba-140	4.80E-02
Rb-88	1.49E-02	La-140	4.27E-03
Rb-89	6.18E-04	La-141	8.80E-08
Sr-89	1.67E-04	La-142	1.19E-08
Sr-90	1.43E-05	Ce-141	1.58E-04
Sr-91	6.67E-04	Ce-143	3.25E-04
Sr-92	2.36E-04	Ce-144	2.99E-03
Y-90	1.86E-06	Pr-143	6.93E-05
Y-91	1.25E-04	Pr-144	1.69E-03
Y-91m	2.67E-05	Nd-147	1.07E-06
Y-92	9.01E-04	Np-239	2.99E-02
Y-93	7.25E-04	Pu-238	2.64E-09
Zr-95	2.20E-03	Pu-239	3.39E-10
Zr-97	4.40E-07	Pu-240	4.27E-10
Nb-95	1.07E-03	Pu-241	1.28E-07
Mo-99	4.52E-02	Am-241	1.85E-10
Tc-99m	2.27E-02	Cm-242	3.78E-08
Tc-99	1.76E-08	Cm-244	1.76E-09
Ru-103	2.63E-03	W-187	6.30E-04
		Total	8.87E+02

Attachment 3
Basis of Source Term for Normal Gaseous Radioactive Effluent Release

Guidance from Nuclear Energy Institute (NEI) 10-01 was used to determine the normal gaseous release source terms. The definition of the parameter noted in NEI 10-01 is "The expected annual activity, by radionuclide, contained in routine plant airborne effluent streams." The release activities are used in the Environmental Report (Section 3.5, Radioactive Waste Management System) and the Site Safety Analysis Report (Section 11.3.3, Gaseous Radioactive Releases) to describe the gaseous radioactive waste management subsystem and ensure compliance with 10 CFR 20 and the ALARA principles of 10 CFR Part 50, Appendix I.

In accordance with the guidelines of NEI 10-01, source term information was obtained from each of the four small modular reactor (SMR) vendors for the normal gaseous effluent release activities. The SMR vendors provided the information with various levels of computational maturity. In some instances, values of activity were overly conservative and adjustments were made on a case by case basis.

One vendor provided source terms using methodology that follows the guidance provided in NUREG-0017 with plant specific primary coolant concentration and gas stripper parameters. Although the PWR-GALE code was not used because several built in parameters are not applicable to integrated pressurized water reactors, the underlying methodology was utilized, where applicable, with vendor specific parameters. Tennessee Valley Authority (TVA) increased the activity of each isotope by 10% for conservatism due to the preliminary nature of the analysis.

Another vendor provided values based on a power level ratio for the PSEG early site permit application (ESPA) (Reference 5). The PSEG submittal contains isotopes and activities from both PWR and ABWR designs of various power levels. Included in the PSEG ESPA submittal is a U.S. EPR which has a power level of 4,500 MWth. Evaluation of the activities provided show that several parameters, especially xenon (Xe) and krypton (Kr) noble gases, were found to be overly conservative when compared to the other vendor provided information. This is due primarily to use of ABWR values and conservative scaling factors applied to the U.S. EPR activities. Subsequently, Xe and Kr isotopes from this vendor were considered to be overly conservative and omitted from consideration in the Clinch River Nuclear (CRN) surrogate plant values. Also noted in Table 7-A, the activity values for Kr-85 and Kr-85m were inadvertently reversed as presented in the table. (The correct values are Kr-85 = 633 Ci/yr and Kr-85m = 23.2 Ci/yr.)

A third vendor provided a set of isotopes and activities based on preliminary evaluations utilizing typical methods. Subsequent to the values provided, a revised methodology was utilized by the vendor to determine the source term values. Discussion with the vendor indicated that the revised methodology will result in substantial reductions in the activities associated with various isotopes.

A fourth vendor provided a set of isotopes and activities based on use of PWR-GALE with basic inputs associated with their SMR design. This vendor is currently developing vendor specific software and/or modifications to PWR GALE to more accurately predict gaseous releases. Discussions with the vendor indicated that substantial margin exists in the preliminary values provided. It was noted that the value for Kr-85 was significantly higher than a scaled AP1000. A review of the Bell Bend Combined License Application (COLA) noted that adjustments to the chemical shim bleed flow rate resulted in significant reductions to Kr-85 activity (i.e., a factor of 12) due to code input restrictions (see discussion below). Subsequently, the Kr-85 activity was

considered to be overly conservative and was omitted from the CRN surrogate evaluation. A sensitivity study was performed utilizing the provided values (including Kr-85) for a single unit and for a site. The sensitivity study resulted in doses that are within regulatory limits and are bounded by the doses based on the surrogate plant values. Note that for the sensitivity study, the value for carbon-14 (C-14) was reduced to a value of 3.0 for the unit and 10.0 for the site computation. This is justified by use of C-14 activity levels produced using EPRI Report 1021106 (Reference 1). (See discussion below.)

TVA created a composite table on a per unit basis and on a total site basis. The site basis multiplied the activities of each vendor by the anticipated number of units for the CRN Site (e.g. mPower x 4, NuScale x 12, etc).

The composite table of source terms for the per unit basis and total site basis was then utilized to compute the offsite dose consequences. The analysis utilized site specific atmospheric dispersion factors and demography information along with standard methodologies to determine if acceptable dose limits were met. The results of the computations indicate that all regulatory acceptance criteria are met.

Table 7-A is a composite table that includes annual activity, by radionuclide, on a per unit basis. Table 7-B is a composite table that includes annual activity, by radionuclide, on a per site basis. During engineering analysis of this parameter, it was identified that some of the radionuclide annual activities contained excessive conservatism in their development. Adjustments were made in some cases such that the annual activity values in the tables are not bounding for a vendor, or multiple vendors. The specific annual activities for the radionuclides not bounded are highlighted in red. This is acceptable due to the limited development of the fuel design for the various vendors. Any variances to the values considered will be evaluated during the development of the COLA.

Evaluation of Source Terms for Reasonableness

Release activities in gaseous effluents are typically evaluated based on the methodology outlined in NUREG-0017. However, the methodology outlined in NUREG-0017 is based, in part, on empirical data generated by large light water reactors (LWRs) from the 1970's. And in several cases, the predicted activities are fixed and/or independent of plant operating practices. With respect to proposed SMRs, plant designs and operating practices will change significantly and will likely result in computational changes in the methods to predict source terms used for normal gaseous and liquid effluent releases. This has been recognized by SMR vendors and the NRC (as noted in public meetings with SMR vendors, Reference 2) and will require evaluation and/or potential modifications to the PWR-GALE code to evaluate technical review of proposed small integrated pressurized water reactor (iPWR) designs.

Until such time that the updated methodologies are accepted by NRC, a comparison of the CRN surrogate plant source terms to previously submitted source terms is made to assess reasonableness for use in evaluation of dose predictions. For normal gaseous effluents TVA compared the release concentrations submitted by Southern Nuclear Operating Company in support of the Vogtle ESPA. The values are provided in Table 11.2.3 of the Vogtle ESPA are for two proposed AP1000 units (Reference 3). A comparison is made in Table 1 on an individual isotope bases by considering a ratio of the AP1000 core power to that of the surrogate plant (3400 Mwth / 800 Mwth). Table 1 compares the CRN surrogate isotopes on a

unit basis to a scaled down AP1000 and depicts the relative differences between the two sets of data. The CRN surrogate values that are lower than the scaled down AP1000 are highlighted.

With only a few exceptions, the CRN surrogate unit utilizes source terms that are somewhat larger than the scaled AP1000 source terms and are within the expected range for this type of comparison. Note that for isotopes with very low activities, a greater proportion of difference may be noted due to relative uncertainties in computation methods. A few of the activities for CRN values are lower than the scaled AP1000 values. The noted outliers are Kr-85 (~8 times lower), Xe-131m (~ 35% lower), Xe-133 (~ 50% lower) Xe-135 (10% lower), and antimony (Sb-125) (~33% lower). These outliers are highlighted on Table 1.

A review of the Bell Bend COLA (Reference 4) identifies similar differences in the values for Kr-85, Xe-131m, and Xe-133. The Bell Bend COLA values are based on the U.S. EPR design, but included a reduction to the total shim bleed flow rate to the liquid waste system for processing to 5%. The PWR GALE code was noted to be limited to 10% shim bleed flow rate. For Bell Bend, reducing the shim bleed flow rate from 10% to 5% reduced the activities of these isotopes similarly to those of the CRN surrogate values. This same effect should also be realized with the CRN surrogate plant which is expected to have similar shim bleed flow rates.

However, because the Kr-85 activity value was a significant outlier for one of the vendors, a sensitivity evaluation was performed (which utilized a higher value than the scaled AP1000 value) to assess the overall impact. This study resulted in acceptable dose consequences for both the unit and site analysis.

The CRN activity values for the remaining outliers Xe-135 and Sb-125 are within expected ranges based on a comparison to a scaled AP1000.

Another noted isotope that is generally different than those predicted by use of PWR-GALE is C-14. This isotope is a fixed value in the current version of PWR-GALE and is expected to overpredict the values for large PWRs and subsequently, an SMR. The PWR-GALE value predicted for C-14 has been noted to generally overpredict the activity due to improvements in plant operating conditions over the past years. EPRI Report 1021106 (Reference 1) provides alternate means for predicting the activity of C-14 based on reactor power levels. Figure 4-4 of the report notes that a proxy generation of C-14 for PWRs will approximate 3.4 Ci/GWth-yr for Westinghouse units. Using the proxy method for the CRN surrogate plant yields ~ 2.7 Ci/yr (3.4 x 0.8) on a per unit basis (CRN = 800 MW) or 8.16 Ci/yr (3.4 x 2.4) for the site (CRN = 2,400 MW). TVA's analysis used values in excess of this value for both evaluations. Subsequently, the activities for normal gaseous effluents are not unreasonable.

Basis for Surrogate Plant Representing Vendor Provided Information

TVA also evaluated all vendor provided information to provide assurance that each would be bounded by the CRN surrogate plant on a unit basis and a site basis.

For two of the vendors, the CRN surrogate plant activities on a per unit basis and a site basis bound the information provided for all isotopes. (See Tables 7-A and 7-B.)

As discussed above, the values provided by one vendor are based on a ratio of power levels to the PSEG ESPA. The PSEG ESPA is based on values predicted for various plant types and sizes and is overly conservative due to the design method used. However, the values provided by one of the vendors was increased by 10% for discretionary conservatism. The additional margin incorporated for this vendor will adequately serve as a proxy.

Note that subsequent information provided by one of the vendors demonstrates that argon-41 (Ar-41) production will be less than the average for large LWRs and will be also be bounded by the surrogate releases. This conclusion resulted from further advancements in the vendor fuel and waste management design and analysis.

For one of the vendors, the CRN surrogate plant activities on a per unit basis bound the information provided for all isotopes with the exception of Kr-85 and C-14. However, as provided in discussions above, C-14 is overly conservative for SMR designs as presented in Reference 1. Additionally, a sensitivity study was performed using the values provided to ensure the dose impacts for these activities were acceptable on both a per unit basis and a site basis. Included in the sensitivity study was an adjustment for C-14 based on the methodology described in Reference 1.

Based on this review, TVA expects all SMR vendors to be bounded by the CRN surrogate plant values for gaseous effluent release.

References

1. Estimation of Carbon-14 in Nuclear Power Plant Gaseous Effluents, Electric Power Research Institute, Technical Report 1021106, Final Report 2010.

2. NRC Meeting Notes, "Review of the PWR-GALE Code (GALE86), Potential Code Features Requiring Modifications for Evaluating iPWR Applications," ADAMS Accession number ML12173A102.

3. Southern Nuclear Operating Company, Vogtle Early Site Permit Application, Part 2 - Site Safety Analysis Report, Chapter 11, ADAMS Accession number ML091540891.

4. Bell Bend COLA, Chapter 11 - Radioactive Waste Management, ADAMS Accession number ML082890654.

5. PSEG Site ESP Application Part 2, Site Safety Analysis Report, Chapter 1 and Chapter 11, ADAMS Accession numbers ML15169A281 and ML15169A284.

Table 1 - Comparison of CRN Surrogate Values to Scaled AP1000

Radionuclide	Surrogate Release 1 unit Ci/yr	AP1000 Release 2 units Ci/yr	AP1000 Release 1 unit Ci/yr	AP1000 Scaled to CRN Surrogate Ci/yr	Delta (CRN Surrogate - Scaled AP1000)	Ratio (CRN Surrogate / Scaled AP1000)
Kr-83m	1.07E-03					
Kr-85m	8.47E+01	7.2E+01	3.6E+01	8.5E+00	7.6E+01	10.00
Kr-85	1.21E+02	8.2E+03	4.1E+03	9.6E+02	-8.4E+02	0.13
Kr-87	8.18E+00	3.0E+01	1.5E+01	3.5E+00	4.7E+00	2.32
Kr-88	3.63E+01	9.2E+01	4.6E+01	1.1E+01	2.5E+01	3.35
Kr-89	1.25E-07					
Kr-90						
Xe-131m	2.75E+02	3.6E+03	1.8E+03	4.2E+02	-1.5E+02	0.65
Xe-133m	2.63E+01	1.7E+02	8.5E+01	2.0E+01	6.3E+00	1.31
Xe-133	5.61E+02	9.2E+03	4.6E+03	1.1E+03	-5.2E+02	0.52
Xe-135m	3.19E+00	1.4E+01	7.0E+00	1.6E+00	1.5E+00	1.94
Xe-135	7.04E+01	6.6E+02	3.3E+02	7.8E+01	-7.2E+00	0.91
Xe-137	7.50E-01					
Xe-138	2.86E+00	1.2E+01	6.0E+00	1.4E+00	1.4E+00	2.03
Xe-139						
I-129	6.68E-12					
I-131	7.70E-02	2.4E-01	1.2E-01	2.8E-02	4.9E-02	2.73
I-132	3.38E-01					
I-133	2.63E-01	8.0E-01	4.0E-01	9.4E-02	1.7E-01	2.79
I-134	5.84E-01					
I-135	3.72E-01					
H-3	3.10E+02	7.0E+02	3.5E+02	8.2E+01	2.3E+02	3.76
C-14	7.30E+00	1.5E+01	7.5E+00	1.8E+00	5.5E+00	4.14
Na-24	6.25E-04					
P-32	1.42E-04					
Ar-41	4.00E+01	6.8E+01	3.4E+01	8.0E+00	3.2E+01	5.00
Cr-51	5.42E-03	1.2E-03	6.0E-04	1.4E-04	5.3E-03	38.39
Mn-54	8.35E-04	8.6E-04	4.3E-04	1.0E-04	7.3E-04	8.26
Mn-56	5.42E-04					
Fe-55	1.00E-03					
Fe-59	1.25E-04	1.6E-04	8.0E-05	1.9E-05	1.1E-04	6.65
Co-58	2.30E-02	4.6E-02	2.3E-02	5.4E-03	1.8E-02	4.25
Co-60	8.80E-03	1.7E-02	8.5E-03	2.0E-03	6.8E-03	4.40
Ni-63	1.22E-03					
Cu-64	1.54E-03					
Zn-65	1.71E-03					
Br-84	1.07E-06					
Rb-88	8.17E-07					
Rb-89	6.67E-06					
Sr-89	3.00E-03	6.0E-03	3.0E-03	7.1E-04	2.3E-03	4.25
Sr-90	1.20E-03	2.4E-03	1.2E-03	2.8E-04	9.2E-04	4.25
Y-90	7.09E-06					
Sr-91	1.54E-04					
Sr-92	1.21E-04					
Y-91	3.72E-05					
Y-92	9.60E-05					
Y-93	1.71E-04					
Zr-95	1.00E-03	2.0E-03	1.0E-03	2.4E-04	7.6E-04	4.25
Nb-95	2.50E-03	5.0E-03	2.5E-03	5.9E-04	1.9E-03	4.25
Mo-99	9.19E-03					
Tc-99m	4.59E-05					

Attachment 3
Basis of Source Term for Normal Gaseous Radioactive Effluent Release

Radionuclide	Surrogate Release 1 unit Ci/yr	AP1000 Release 2 units Ci/yr	AP1000 Release 1 unit Ci/yr	AP1000 Scaled to CRN Surrogate Ci/yr	Delta (CRN Surrogate - Scaled AP1000)	Ratio (CRN Surrogate / Scaled AP1000)
Ru-103	5.42E-04	1.6E-04	8.0E-05	1.9E-05	5.2E-04	28.79
Rh-103m	1.23E-09					
Ru-106	7.80E-05	1.6E-04	8.0E-05	1.9E-05	5.9E-05	4.14
Rh-106	3.81E-12					
Ag-110m	1.78E-04					
Sb-124	2.79E-05					
Te-129m	3.38E-05					
Te-131m	1.17E-05					
Te-132	5.94E-06					
Cs-134	2.30E-03	4.6E-03	2.3E-03	5.4E-04	1.8E-03	4.25
Cs-136	9.19E-05	1.7E-04	8.5E-05	2.0E-05	7.2E-05	4.59
Cs-137	8.14E-03	7.2E-03	3.6E-03	8.5E-04	7.3E-03	9.61
Cs-138	2.63E-05					
Ba-140	4.17E-03	8.4E-04	4.2E-04	9.9E-05	4.1E-03	42.19
La-140	2.79E-04					
Ce-141	1.42E-03	8.4E-05	4.2E-05	9.9E-06	1.4E-03	143.59
Ce-143	9.63E-09					
Ce-144	2.92E-06					
Pr-144	2.92E-06					
W-187	2.92E-05					
Np-239	1.84E-03					
Sb-125	9.42E-06	1.2E-04	6.0E-05	1.4E-05	-4.7E-06	0.67
Co-57	2.75E-05					
Total	1.55E+03	2.3E+04	1.1E+04	2.7E+03		

Basis of Source Term for Normal Gaseous Radioactive Effluent Release

Vendor 1

TABLE 7 -A
One Unit
Average Annual Normal Gaseous Radioactive Release

Radionuclide	Release 1 unit Ci/y	Radionuclide	Release 1 unit Ci/y
Kr-85m	8.47E+01	Ar-41	1.98E+00
Kr-85	1.21E+02	Cr-51	4.95E-04
Kr-87	7.15E+00	Mn-54	8.80E-05
Kr-88	3.63E+01	Co-57	2.75E-05
Xe-131m	2.75E+02	Co-58	7.15E-03
Xe-133m	3.96E+00	Co-60	9.68E-04
Xe-133	5.61E+02	Fe-59	1.98E-07
Xe-135m	3.19E+00	Sr-89	4.07E-06
Xe-135	7.04E+01	Sr-90	2.09E-04
Xe-137	6.49E-01	Zr-95	3.30E-04
Xe-138	2.86E+00	Nb-95	1.65E-07
I-131	8.03E-03	Ru-106	8.69E-06
I-132	4.95E-03	Sb-125	1.21E-08
I-133	1.43E-02	Cs-134	1.21E-08
I-134	3.63E-03	Cs-137	8.14E-03
I-135	1.00E-02	Ba-140	2.20E-06
H-3	2.20E+01	Ce-141	8.36E-07
C-14	2.42E+00	Total	1.19E+03

Vendor 2

TABLE 7 -A
One Unit
Average Annual Normal Gaseous Radioactive Release

	Radionuclide	Release 1 unit Ci/y	Radionuclide	Release 1 unit Ci/y
	Kr-83m	1.29E-04	Rb-88	NA
**	Kr-85m	6.33E+02	Rb-89	6.67E-06
	Kr-85	2.32E+01	Sr-89	8.77E-04
	Kr-87	8.18E+00	Sr-90	1.85E-04
	Kr-88	2.78E+01	Y-90	7.09E-06
**	Kr-89	3.72E+01	Sr-91	1.54E-04
	Kr-90	NA	Sr-92	1.21E-04
**	Xe-131m	4.17E+02	Y-91	3.72E-05
	Xe-133m	2.63E+01	Y-92	9.60E-05
**	Xe-133	1.11E+03	Y-93	1.71E-04
**	Xe-135m	6.25E+01	Zr-95	2.46E-04
**	Xe-135	1.85E+02	Nb-95	1.29E-03
**	Xe-137	7.94E+01	Mo-99	9.19E-03
**	Xe-138	6.67E+01	Tc-99m	4.59E-05
	Xe-139	NA	Ru-103	5.42E-04
	I-129	NA	Rh-103m	NA
	I-131	4.00E-02	Ru-106	1.20E-05
	I-132	3.38E-01	Rh-106	NA
	I-133	2.63E-01	Ag-110m	3.09E-07
	I-134	5.84E-01	Sb-124	2.79E-05
	I-135	3.72E-01	Te-129m	3.38E-05
	H-3	5.40E+01	Te-131m	1.17E-05
	C-14	2.92E+00	Te-132	2.92E-06
	Na-24	6.25E-04	Cs-134	9.60E-04
	P-32	1.42E-04	Cs-136	9.19E-05
	Ar-41	5.25E+00	Cs-137	1.46E-03
	Cr-51	5.42E-03	Cs-138	2.63E-05
	Mn-54	8.35E-04	Ba-140	4.17E-03
	Mn-56	5.42E-04	La-140	2.79E-04
	Fe-55	1.00E-03	Ce-141	1.42E-03
	Fe-59	1.25E-04	Ce-143	NA
	Co-58	3.55E-03	Ce-144	2.92E-06
	Co-60	2.01E-03	Pr-144	2.92E-06
	Ni-63	1.00E-06	W-187	2.92E-05
	Cu-64	1.54E-03	Np-239	1.84E-03
	Zn-65	1.71E-03	Co-57	1.27E-06
	Br-84	NA	Sb-125	9.42E-06
			Total	2.74E+03

Vendor 3

TABLE 7 -A
One Unit
Average Annual Normal Gaseous Radioactive Release

	Radionuclide	Release 1 unit Ci/y	Radionuclide	Release 1 unit Ci/y
	Kr-83m	1.07E-03	Rb-88	8.17E-07
	Kr-85m	5.34E-03	Rb-89	8.50E-08
	Kr-85	2.47E-01	Sr-89	4.87E-08
	Kr-87	1.96E-03	Sr-90	4.33E-09
	Kr-88	7.77E-03	Y-90	6.37E-10
	Kr-89	3.92E-10	Sr-91	3.29E-08
	Kr-90		Sr-92	1.37E-08
	Xe-131m	1.05E-01	Y-91	9.99E-09
	Xe-133m	4.85E-02	Y-92	3.43E-09
	Xe-133	9.84E+00	Y-93	6.51E-09
	Xe-135m	2.21E-04	Zr-95	1.47E-05
	Xe-135	5.87E-02	Nb-95	1.30E-08
	Xe-137	9.83E-09	Mo-99	3.09E-04
	Xe-138	9.88E-05	Tc-99m	7.00E-06
	Xe-139		Ru-103	1.06E-08
	I-129	6.68E-12	Rh-103m	1.23E-09
	I-131	2.19E-04	Ru-106	3.26E-09
	I-132	8.08E-05	Rh-106	3.81E-12
	I-133	3.47E-04	Ag-110m	1.78E-04
	I-134	4.21E-05	Sb-124	
	I-135	1.95E-04	Te-129m	5.33E-10
	H-3	8.42E+01	Te-131m	4.75E-07
	C-14	3.11E-01	Te-132	5.94E-06
	Na-24		Cs-134	5.80E-06
	P-32		Cs-136	1.83E-06
*	Ar-41	1.32E+02	Cs-137	7.15E-06
	Cr-51	8.57E-04	Cs-138	1.96E-06
	Mn-54	4.35E-04	Ba-140	8.07E-08
	Mn-56		La-140	1.15E-08
	Fe-55	3.24E-04	Ce-141	1.24E-08
	Fe-59	7.96E-05	Ce-143	9.63E-09
	Co-58	6.12E-04	Ce-144	9.88E-09
	Co-60	6.12E-05	Pr-144	3.87E-10
	Ni-63	1.22E-03	W-187	
	Cu-64		Np-239	2.04E-04
	Zn-65	1.41E-04		
	Br-84	1.07E-06		
			Total	2.27E+02

Vendor 4

TABLE 7 -A
One Unit
Average Annual Normal Gaseous Radioactive Release

	Radionuclide	Release 1 unit Ci/y	Radionuclide	Release 1 unit Ci/y
	Kr-83m		Rb-88	
	Kr-85m	4.00E+00	Rb-89	
***	Kr-85	1.20E+03	Sr-89	3.00E-03
	Kr-87	4.00E+00	Sr-90	1.20E-03
	Kr-88	7.00E+00	Y-90	
	Kr-89		Sr-91	
	Kr-90		Sr-92	
	Xe-131m	1.20E+02	Y-91	
	Xe-133m	1.00E+00	Y-92	
	Xe-133	1.60E+02	Y-93	
	Xe-135m	3.00E+00	Zr-95	1.00E-03
	Xe-135	2.40E+01	Nb-95	2.50E-03
	Xe-137	0.00E+00	Mo-99	
	Xe-138	2.00E+00	Tc-99m	
	Xe-139		Ru-103	8.00E-05
	I-129		Rh-103m	
	I-131	7.70E-02	Ru-106	7.80E-05
	I-132		Rh-106	
	I-133	2.60E-01	Ag-110m	
	I-134		Sb-124	
	I-135		Te-129m	
	H-3	3.10E+02	Te-131m	
	C-14	7.30E+00	Te-132	
	Na-24		Cs-134	2.30E-03
	P-32		Cs-136	8.50E-05
	Ar-41	3.40E+01	Cs-137	3.60E-03
	Cr-51	6.10E-04	Cs-138	
	Mn-54	4.30E-04	Ba-140	4.20E-04
	Mn-56		La-140	
	Fe-55		Ce-141	4.20E-05
	Fe-59	7.90E-05	Ce-143	
	Co-58	2.30E-02	Ce-144	
	Co-60	8.80E-03	Pr-144	
	Ni-63		W-187	
	Cu-64		Np-239	
	Zn-65			
	Br-84			
			Total	1.88E+03

Composite Table

TABLE 7 - A
One Unit
Average Annual Normal Gaseous Radioactive Release

Radionuclide	Release 1 unit Ci/y	Radionuclide	Release 1 unit Ci/y
Kr-83m	1.07E-03	Rb-88	8.17E-07
Kr-85m	8.47E+01	Rb-89	6.67E-06
Kr-85	1.21E+02	Sr-89	3.00E-03
Kr-87	8.18E+00	Sr-90	1.20E-03
Kr-88	3.63E+01	Y-90	7.09E-06
Kr-89	1.25E-07	Sr-91	1.54E-04
Kr-90		Sr-92	1.21E-04
Xe-131m	2.75E+02	Y-91	3.72E-05
Xe-133m	2.63E+01	Y-92	9.60E-05
Xe-133	5.61E+02	Y-93	1.71E-04
Xe-135m	3.19E+00	Zr-95	1.00E-03
Xe-135	7.04E+01	Nb-95	2.50E-03
Xe-137	7.50E-01	Mo-99	9.19E-03
Xe-138	2.86E+00	Tc-99m	4.59E-05
Xe-139		Ru-103	5.42E-04
I-129	6.68E-12	Rh-103m	1.23E-09
I-131	7.70E-02	Ru-106	7.80E-05
I-132	3.38E-01	Rh-106	3.81E-12
I-133	2.63E-01	Ag-110m	1.78E-04
I-134	5.84E-01	Sb-124	2.79E-05
I-135	3.72E-01	Te-129m	3.38E-05
H-3	3.10E+02	Te-131m	1.17E-05
C-14	7.30E+00	Te-132	5.94E-06
Na-24	6.25E-04	Cs-134	2.30E-03
P-32	1.42E-04	Cs-136	9.19E-05
Ar-41	4.00E+01	Cs-137	8.14E-03
Cr-51	5.42E-03	Cs-138	2.63E-05
Mn-54	8.35E-04	Ba-140	4.17E-03
Mn-56	5.42E-04	La-140	2.79E-04
Fe-55	1.00E-03	Ce-141	1.42E-03
Fe-59	1.25E-04	Ce-143	9.63E-09
Co-58	2.30E-02	Ce-144	2.92E-06
Co-60	8.80E-03	Pr-144	2.92E-06
Ni-63	1.22E-03	W-187	2.92E-05
Cu-64	1.54E-03	Np-239	1.84E-03
Zn-65	1.71E-03	Sb-125	9.42E-06
Br-84	1.07E-06	Co-57	2.75E-05
		Total	1.55E+03

Notes:
Items highlighted in Yellow are bounding values for surrogate plant (Composite Table)
Items highlighted in Red are not bounded by surrogate plant value (Composite Table)

* Ar-41 Adjusted to due excessive conservatism
** Values Excluded Due to Excessive Conservatism
*** Value used in Sensitivity Study

Attachment 3
Basis of Source Term for Normal Gaseous Radioactive Effluent Release

Vendor 1

TABLE 7 - B For Site Average Annual Normal Gaseous Radioactive Release			
Radionuclide	Release Ci/y	Radionuclide	Release Ci/y
Kr-85m	3.39E+02	Ar-41	7.92E+00
Kr-85	4.84E+02	Cr-51	1.98E-03
Kr-87	2.86E+01	Mn-54	3.52E-04
Kr-88	1.45E+02	Co-57	1.10E-04
Xe-131m	1.10E+03	Co-58	2.86E-02
Xe-133m	1.58E+01	Co-60	3.87E-03
Xe-133	2.24E+03	Fe-59	7.92E-07
Xe-135m	1.28E+01	Sr-89	1.63E-05
Xe-135	2.82E+02	Sr-90	8.36E-04
Xe-137	2.60E+00	Zr-95	1.32E-03
Xe-138	1.14E+01	Nb-95	6.60E-07
I-131	3.21E-02	Ru-106	3.48E-05
I-132	1.98E-02	Sb-125	4.84E-08
I-133	5.72E-02	Cs-134	4.84E-08
I-134	1.45E-02	Cs-137	3.26E-02
I-135	4.00E-02	Ba-140	8.80E-06
H-3	8.80E+01	Ce-141	3.34E-06
C-14	9.68E+00		
		Total	4.77E+03

Vendor 2

TABLE 7 -B For Site Average Annual Normal Gaseous Radioactive Release			
Radionuclide	Release Ci/y	Radionuclide	Release Ci/y
Kr-83m	5.18E-04	Rb-88	NA
Kr-85m	2.53E+03	Rb-89	2.67E-05
Kr-85	9.26E+01	Sr-89	3.51E-03
Kr-87	3.27E+01	Sr-90	7.41E-04
Kr-88	1.11E+02	Y-90	2.84E-05
Kr-89	1.49E+02	Sr-91	6.18E-04
Kr-90	NA	Sr-92	4.84E-04
Xe-131m	1.67E+03	Y-91	1.49E-04
Xe-133m	1.05E+02	Y-92	3.84E-04
Xe-133	4.45E+03	Y-93	6.86E-04
Xe-135m	2.50E+02	Zr-95	9.82E-04
Xe-135	7.41E+02	Nb-95	5.18E-03
Xe-137	3.17E+02	Mo-99	3.68E-02
Xe-138	2.67E+02	Tc-99m	1.83E-04
Xe-139	NA	Ru-103	2.17E-03
I-129	NA	Rh-103m	NA
I-131	1.60E-01	Ru-106	4.82E-05
I-132	1.35E+00	Rh-106	NA
I-133	1.05E+00	Ag-110m	1.24E-06
I-134	2.33E+00	Sb-124	1.12E-04
I-135	1.49E+00	Te-129m	1.35E-04
H-3	2.16E+02	Te-131m	4.68E-05
C-14	1.17E+01	Te-132	1.17E-05
Na-24	2.50E-03	Cs-134	3.84E-03
P-32	5.68E-04	Cs-136	3.68E-04
Ar-41	2.10E+01	Cs-137	5.84E-03
Cr-51	2.17E-02	Cs-138	1.05E-04
Mn-54	3.34E-03	Ba-140	1.67E-02
Mn-56	2.17E-03	La-140	1.12E-03
Fe-55	4.01E-03	Ce-141	5.68E-03
Fe-59	5.01E-04	Ce-143	NA
Co-58	1.42E-02	Ce-144	1.17E-05
Co-60	8.03E-03	Pr-144	1.17E-05
Ni-63	4.01E-06	W-187	1.17E-04
Cu-64	6.18E-03	Np-239	7.35E-03
Zn-65	6.86E-03	Co-57	5.06E-06
Br-84	NA	Sb-125	3.77E-05
		Total	1.10E+04

Vendor 3

TABLE 7 -B For Site Average Annual Normal Gaseous Radioactive Release			
Radionuclide	Release Ci/y	Radionuclide	Release Ci/y
Kr-83m	1.28E-02	Rb-88	9.80E-06
Kr-85m	6.41E-02	Rb-89	1.02E-06
Kr-85	2.96E+00	Sr-89	5.84E-07
Kr-87	2.35E-02	Sr-90	5.20E-08
Kr-88	9.32E-02	Y-90	7.64E-09
Kr-89	4.70E-09	Sr-91	3.95E-07
Kr-90		Sr-92	1.64E-07
Xe-131m	1.26E+00	Y-91	1.20E-07
Xe-133m	5.82E-01	Y-92	4.12E-08
Xe-133	1.18E+02	Y-93	7.81E-08
Xe-135m	2.65E-03	Zr-95	1.76E-04
Xe-135	7.04E-01	Nb-95	1.56E-07
Xe-137	1.18E-07	Mo-99	3.71E-03
Xe-138	1.19E-03	Tc-99m	8.40E-05
Xe-139		Ru-103	1.27E-07
I-129	8.02E-11	Rh-103m	1.48E-08
I-131	2.63E-03	Ru-106	3.91E-08
I-132	9.70E-04	Rh-106	4.57E-11
I-133	4.16E-03	Ag-110m	2.14E-03
I-134	5.05E-04	Sb-124	
I-135	2.34E-03	Te-129m	6.40E-09
H-3	1.01E+03	Te-131m	5.70E-06
C-14	3.73E+00	Te-132	7.13E-05
Na-24		Cs-134	6.96E-05
P-32		Cs-136	2.20E-05
Ar-41	1.58E+03	Cs-137	8.58E-05
Cr-51	1.03E-02	Cs-138	2.35E-05
Mn-54	5.22E-03	Ba-140	9.68E-07
Mn-56		La-140	1.38E-07
Fe-55	3.89E-03	Ce-141	1.49E-07
Fe-59	9.55E-04	Ce-143	1.16E-07
Co-58	7.34E-03	Ce-144	1.19E-07
Co-60	7.34E-04	Pr-144	4.64E-09
Ni-63	1.46E-02	W-187	
Cu-64		Np-239	2.45E-03
Zn-65	1.69E-03		
Br-84	1.28E-05		
		Total	2.72E+03

Vendor 4

TABLE 7 -B For Site Average Annual Normal Gaseous Radioactive Release			
Radionuclide	Release Ci/y	Radionuclide	Release Ci/y
Kr-83m		Rb-88	
Kr-85m	1.20E+01	Rb-89	
Kr-85	3.60E+03	Sr-89	9.00E-03
Kr-87	1.20E+01	Sr-90	3.60E-03
Kr-88	2.10E+01	Y-90	
Kr-89		Sr-91	
Kr-90		Sr-92	
Xe-131m	3.60E+02	Y-91	
Xe-133m	3.00E+00	Y-92	
Xe-133	4.80E+02	Y-93	
Xe-135m	9.00E+00	Zr-95	3.00E-03
Xe-135	7.20E+01	Nb-95	7.50E-03
Xe-137		Mo-99	
Xe-138	6.00E+00	Tc-99m	
Xe-139		Ru-103	2.40E-04
I-129		Rh-103m	
I-131	2.31E-01	Ru-106	2.34E-04
I-132		Rh-106	
I-133	7.80E-01	Ag-110m	
I-134		Sb-124	
I-135		Te-129m	
H-3	9.30E+02	Te-131m	
C-14	2.19E+01	Te-132	
Na-24		Cs-134	6.90E-03
P-32		Cs-136	2.55E-04
Ar-41	1.02E+02	Cs-137	1.08E-02
Cr-51	1.83E-03	Cs-138	
Mn-54	1.29E-03	Ba-140	1.26E-03
Mn-56		La-140	
Fe-55		Ce-141	1.26E-04
Fe-59	2.37E-04	Ce-143	
Co-58	6.90E-02	Ce-144	
Co-60	2.64E-02	Pr-144	
Ni-63		W-187	
Cu-64		Np-239	
Zn-65			
Br-84			
		Total	5.63E+03

Composite Table

TABLE 7 - B For Site Average Annual Normal Gaseous Radioactive Release			
Radionuclide	Release Ci/y	Radionuclide	Release Ci/y
Kr-83m	1.28E-02	Rb-88	9.80E-06
Kr-85m	3.39E+02	Rb-89	2.67E-05
Kr-85	7.20E+02	Sr-89	9.00E-03
Kr-87	3.27E+01	Sr-90	3.60E-03
Kr-88	1.45E+02	Y-90	2.84E-05
Kr-89	5.00E-07	Sr-91	6.18E-04
Kr-90		Sr-92	4.84E-04
Xe-131m	1.67E+03	Y-91	1.49E-04
Xe-133m	1.05E+02	Y-92	3.84E-04
Xe-133	2.24E+03	Y-93	6.86E-04
Xe-135m	1.28E+01	Zr-95	3.00E-03
Xe-135	2.82E+02	Nb-95	7.50E-03
Xe-137	3.00E+00	Mo-99	3.68E-02
Xe-138	1.14E+01	Tc-99m	1.83E-04
Xe-139		Ru-103	2.17E-03
I-129	8.02E-11	Rh-103m	1.48E-08
I-131	2.31E-01	Ru-106	2.34E-04
I-132	1.35E+00	Rh-106	4.57E-11
I-133	1.05E+00	Ag-110m	2.14E-03
I-134	2.33E+00	Sb-124	1.12E-04
I-135	1.49E+00	Te-129m	1.35E-04
H-3	1.01E+03	Te-131m	4.68E-05
C-14	1.00E+01	Te-132	7.13E-05
Na-24	2.50E-03	Cs-134	6.90E-03
P-32	5.68E-04	Cs-136	3.68E-04
Ar-41	5.44E+02	Cs-137	3.26E-02
Cr-51	2.17E-02	Cs-138	1.05E-04
Mn-54	5.22E-03	Ba-140	1.67E-02
Mn-56	2.17E-03	La-140	1.12E-03
Fe-55	4.01E-03	Ce-141	5.68E-03
Fe-59	9.55E-04	Ce-143	1.16E-07
Co-58	6.90E-02	Ce-144	1.17E-05
Co-60	2.64E-02	Pr-144	1.17E-05
Ni-63	1.46E-02	W-187	1.17E-04
Cu-64	6.18E-03	Np-239	7.35E-03
Zn-65	6.86E-03	Sb-125	3.77E-05
Br-84	1.28E-05	Co-57	1.10E-04
		Total	7.13E+03

Notes:
Items highlighted in Yellow are bounding values for surrogate plant (Composite Table)
Items highlighted in Red are not bounded by surrogate plant value (Composite Table)

* C-14 Value Based on EPRI Report 1021106
** Values Excluded Due to Excessive Conservatism
*** Value Used in Senstivity Study

Attachment 4
Basis of Source Term for Accident Liquid Radioactive Effluent Release

Guidance from Nuclear Energy Institute (NEI) 10-01 was used to determine the accidental liquid release source terms. The definition of the parameter noted in NEI 10-01 is "The assumed activity, by radionuclide, contained in accidental liquid radwaste release." The release activities are needed in Site Safety Analysis Report, Subsection 2.4.13, Accidental Releases of Liquid Effluents to Ground and Surface Waters, to evaluate the doses on the public during a postulated accident at the plant and to ensure 10 CFR 20.1301 limits will be met. The parameter is utilized as the initial source term held inside the tank prior to an assumed rupture of the tank. Once the tank rupture is postulated, the contents are released into the environment and analyzed in accordance with standard regulatory methods.

In accordance with the guidelines of NEI 10-01, source term information was obtained for the accident liquid release activities. The four small modular reactor (SMR) vendors provided the information with various levels of computational maturity noted.

Two vendor designs are based on features that mitigate the possibility of a tank failure release. Subsequently, the values provided by these vendors were omitted from further evaluation. One of the vendors supplied values for each isotope as a concentration of activity, in micro-curies per cubic centimeter (µCi/cc).

Another vendor supplied values for each isotope as an activity, in curies (Ci), with an associated tank volume for liquid radioactive waste decay storage. The associated tank volume was 10,000 gallons for liquid radioactive decay storage. The supplied tank size is used for the determination of activity concentrations. These values were converted to a concentration of activity, using the equation below.

$$\frac{[Activity\ in\ Ci] \times \frac{1000000 \mu Ci}{1 Ci}}{[Tank\ Volume\ in\ Gallons] \times \frac{3785.41\ cc}{1\ Gallon}} = [Activity\ Concentration\ in\ \mu Ci/cc]$$

The comparison of the two vendor supplied values, as shown in Table 11-A, shows that concentrations for one vendor are mostly greater (i.e., more conservative) than the other. The activities provided by one vendor are based on unfiltered reactor coolant and are considered very conservative. Because of the large amount of conservatism included, a composite table for this parameter was not utilized as suggested by NEI 10-01. The more conservative vendor values were used as the plant parameter envelope (PPE) values for the surrogate plant. The conservatism considered by the vendors is a reflection of the preliminary design of the waste management systems at the time the information was provided.

During engineering analysis of this parameter, it was identified that some of the radionuclide activities contained excessive conservatism. Zirconium-95 (Zr-95) and niobium-95 (Nb-95) were considered unnecessarily conservative and were reduced to develop a more appropriate table of accident liquid source terms. The activities for Zr-95 and Nb-95 were both reduced to 1.32E+2 µCi/cc, which remains very conservative to the comparison vendor. It is acceptable to maintain a significant amount of margin in these values due to the limited development of the fuel and waste management design for the vendors. Any non-conservative variances to these concentrations will be evaluated during the development of the combined license application (COLA).

TABLE 11-A, Comparison of Vendor Isotopic Concentrations

Radionuclide	Vendor A (μCi/cc)	Vendor B (μCi/cc)	Δ (Vendor A-Vendor B) (μCi/cc)
I-129	1.56E-04	4.60E-08	1.56E-04
I-130	6.41E+01	5.00E-02	6.40E+01
I-131	3.75E+03	7.40E-01	3.75E+03
I-132	5.47E+03	3.70E-01	5.47E+03
I-133	7.71E+03	1.30E+00	7.71E+03
I-134	8.66E+03	2.40E-01	8.66E+03
I-135	7.35E+03	7.90E-01	7.35E+03
H-3	3.30E+00	3.50E+00	-1.95E-01
C-14	3.62E-03	N/A	3.62E-03
Na-24	2.03E+00	3.70E-02	1.99E+00
P-32	5.16E-01	2.07E-03	5.14E-01
Cr-51	2.12E+02	8.11E-02	2.12E+02
Mn-54	1.98E+01	1.23E-03	1.98E+01
Mn-56	5.96E+02	4.00E-02	5.96E+02
Co-58	3.20E+01	3.30E-03	3.20E+01
Co-60	6.85E+00	7.21E-03	6.85E+00
Fe-55	7.90E+01	1.80E-02	7.90E+01
Fe-59	5.10E+00	4.50E-04	5.10E+00
Ni-63	2.54E+00	1.80E-05	2.54E+00
Cu-64	1.39E-02	1.92E-02	-5.33E-03
Zn-65	2.25E-06	3.60E-03	-3.60E-03
Rb-89	3.43E+03	6.90E-02	3.43E+03
Sr-89	3.55E+03	1.53E-03	3.55E+03
Sr-90	4.95E+02	1.26E-04	4.95E+02
Y-90	5.13E+02	1.80E-04	5.13E+02
Sr-91	4.53E+03	2.10E-03	4.53E+03
Y-91	4.66E+03	6.01E-04	4.66E+03
Sr-92	4.85E+03	1.59E-03	4.85E+03
Y-92	4.91E+03	1.23E-03	4.91E+03
Y-93	5.54E+03	2.04E-03	5.54E+03
Zr-95	6.58E+03	1.60E-04	6.58E+03
Nb-95	6.60E+03	1.80E-04	6.60E+03
Mo-99	7.01E+03	2.10E-01	7.01E+03
Tc-99m	6.21E+03	1.10E-01	6.21E+03
Ru-103	5.91E+03	2.91E-04	5.91E+03

Radionuclide	Vendor A (µCi/cc)	Vendor B (µCi/cc)	Δ (Vendor A-Vendor B) (µCi/cc)
Rh-103m	5.90E+03	2.91E-04	5.90E+03
Ru-106	2.28E+03	5.41E-05	2.28E+03
Rh-106	2.42E+03	5.41E-05	2.42E+03
Ag-110m	1.14E+01	1.74E-05	1.14E+01
Sb-124	3.11E+00	N/A	3.11E+00
Te-129m	1.90E+02	2.50E-03	1.90E+02
Te-131m	7.22E+02	6.30E-03	7.22E+02
Te-132	5.35E+03	7.00E-02	5.35E+03
Cs-134	7.96E+02	2.00E+00	7.94E+02
Cs-136	2.65E+02	1.00E+00	2.64E+02
Cs-137	6.47E+02	1.20E+00	6.46E+02
Cs-138	7.17E+03	3.70E-01	7.17E+03
Ba-140	6.61E+03	3.90E-03	6.61E+03
La-140	6.81E+03	3.90E-03	6.81E+03
Ce-141	6.24E+03	4.20E-04	6.24E+03
Ce-144	5.35E+03	1.20E-04	5.35E+03
Pr-143	5.68E+03	8.80E-05	5.68E+03
Np-239	7.19E+04	2.04E-02	7.19E+04

Evaluation of Source Terms for Reasonableness

To demonstrate that the accident liquid source terms developed for the Clinch River Nuclear (CRN) Early Site Permit Application (ESPA) are not unreasonable, Tennessee Valley Authority (TVA) compared activity concentrations and activities to the PSEG ESPA values for accident liquid source terms. Comparison to the PSEG values is appropriate because SMR waste management system designs are not expected to significantly change from existing industry designs.

The PSEG ESPA considers four large light water reactor technologies, consisting of a General Electric Nuclear Energy Advanced Boiling Water Reactor, Westinghouse Electric Company Advanced Passive 1000 (AP1000), AREVA NP, Inc. U.S. EPR, and the Mitsubishi Heavy Industries, Ltd. U.S. Advanced Pressurized-Water Reactor (US-APWR). The methodology used in the PSEG ESPA to determine accident liquid source terms is a composite table developed from the four reactor technologies considered. This composite table methodology is considered conservative as no single vendor contains the highest source terms for all radionuclides.

The PSEG accident liquid source terms included liquid radioactive waste decay storage tank volumes for the technologies considered. The smallest tank volume for any technology considered in the PSEG ESPA, 19,600 gallons, was used to convert the PSEG accident liquid source term activity concentration to activities. The smallest tank volume was used with the

highest activity concentrations. The equation below was used to convert activity concentrations presented in the PSEG accident liquid source term composite table.

$$\frac{[Activity\ Concentration\ in\ \mu Ci/cc]\times[Tank\ Volume\ in\ Gallons]\times\frac{3785.41cc}{1\ Gallon}}{\frac{1000000\mu Ci}{1\ Ci}} = [Activity\ in\ Ci]$$

Comparison of the CRN and PSEG activities and activity concentrations (Table 11-B) shows that most of the CRN activities and activity concentrations are larger (i.e., more conservative) than those determined for the PSEG ESPA. Three surrogate radionuclides are less than PSEG values (tritium [H-3], copper-64 [Cu-64], and zirconium-65 [Zn-65]). The dose impact from these isotopes are minimal due to the conservative methodology used for the activities for the other isotopes.

Note that the adjusted values for Zr-95 and Nb-95 significantly exceed the values determined for the PSEG ESPA. The comparison of CRN and PSEG activities and activity concentrations demonstrates that the CRN values are not unreasonable and that adjustments made to Zr-95 and Nb-95 are not unreasonable.

Table 11-B, One Unit - Accidental Liquid Radioactive Release

Radionuclide	Surrogate	Surrogate	PSEG	PSEG	Δ (Surrogate-PSEG)	Δ (Surrogate-PSEG)
Units	**Ci**	**µCi/cc**	**µCi/cc**	**Ci**	**µCi/cc**	**Ci**
I-129	5.89E-03	1.56E-04	4.60E-08	3.41E-06	1.56E-04	5.89E-03
I-130	2.42E+03	6.41E+01	5.00E-02	3.71E+00	6.40E+01	2.42E+03
I-131	1.42E+05	3.75E+03	7.40E-01	5.49E+01	3.75E+03	1.42E+05
I-132	2.07E+05	5.47E+03	3.70E-01	2.75E+01	5.47E+03	2.07E+05
I-133	2.92E+05	7.71E+03	1.30E+00	9.65E+01	7.71E+03	2.92E+05
I-134	3.28E+05	8.66E+03	2.40E-01	1.78E+01	8.66E+03	3.28E+05
I-135	2.78E+05	7.35E+03	7.90E-01	5.86E+01	7.35E+03	2.78E+05
H-3	1.25E+02	3.30E+00	3.50E+00	2.60E+02	-1.95E-01	-1.35E+02
C-14	1.37E-01	3.62E-03	N/A	0.00E+00	3.62E-03	1.37E-01
Na-24	7.68E+01	2.03E+00	3.70E-02	2.75E+00	1.99E+00	7.41E+01
P-32	1.95E+01	5.16E-01	2.07E-03	1.54E-01	5.14E-01	1.94E+01
Cr-51	8.04E+03	2.12E+02	8.11E-02	6.02E+00	2.12E+02	8.03E+03
Mn-54	7.49E+02	1.98E+01	1.23E-03	9.13E-02	1.98E+01	7.48E+02
Mn-56	2.26E+04	5.96E+02	4.00E-02	2.97E+00	5.96E+02	2.26E+04
Co-58	1.21E+03	3.20E+01	3.30E-03	2.45E-01	3.20E+01	1.21E+03
Co-60	2.59E+02	6.85E+00	7.21E-03	5.35E-01	6.85E+00	2.59E+02
Fe-55	2.99E+03	7.90E+01	1.80E-02	1.34E+00	7.90E+01	2.99E+03
Fe-59	1.93E+02	5.10E+00	4.50E-04	3.34E-02	5.10E+00	1.93E+02
Ni-63	9.63E+01	2.54E+00	1.80E-05	1.34E-03	2.54E+00	9.63E+01

Attachment 4
Basis of Source Term for Accident Liquid Radioactive Effluent Release

Radionuclide	Surrogate	Surrogate	PSEG	PSEG	Δ (Surrogate-PSEG)	Δ (Surrogate-PSEG)
Units	Ci	µCi/cc	µCi/cc	Ci	µCi/cc	Ci
Cu-64	5.25E-01	1.39E-02	1.92E-02	1.42E+00	-5.33E-03	-8.99E-01
Zn-65	8.50E-05	2.25E-06	3.60E-03	2.67E-01	-3.60E-03	-2.67E-01
Rb-89	1.30E+05	3.43E+03	6.90E-02	5.12E+00	3.43E+03	1.30E+05
Sr-89	1.34E+05	3.55E+03	1.53E-03	1.14E-01	3.55E+03	1.34E+05
Sr-90	1.87E+04	4.95E+02	1.26E-04	9.35E-03	4.95E+02	1.87E+04
Y-90	1.94E+04	5.13E+02	1.80E-04	1.34E-02	5.13E+02	1.94E+04
Sr-91	1.71E+05	4.53E+03	2.10E-03	1.56E-01	4.53E+03	1.71E+05
Y-91	1.76E+05	4.66E+03	6.01E-04	4.46E-02	4.66E+03	1.76E+05
Sr-92	1.84E+05	4.85E+03	1.59E-03	1.18E-01	4.85E+03	1.84E+05
Y-92	1.86E+05	4.91E+03	1.23E-03	9.13E-02	4.91E+03	1.86E+05
Y-93	2.10E+05	5.54E+03	2.04E-03	1.51E-01	5.54E+03	2.10E+05
Zr-95	5.00E+03	1.32E+02	1.60E-04	1.19E-02	1.32E+02	5.00E+03
Nb-95	5.00E+03	1.32E+02	1.80E-04	1.34E-02	1.32E+02	5.00E+03
Mo-99	2.65E+05	7.01E+03	2.10E-01	1.56E+01	7.01E+03	2.65E+05
Tc-99m	2.35E+05	6.21E+03	1.10E-01	8.16E+00	6.21E+03	2.35E+05
Ru-103	2.24E+05	5.91E+03	2.91E-04	2.16E-02	5.91E+03	2.24E+05
Rh-103m	2.24E+05	5.90E+03	2.91E-04	2.16E-02	5.90E+03	2.24E+05
Ru-106	8.63E+04	2.28E+03	5.41E-05	4.01E-03	2.28E+03	8.63E+04
Rh-106	9.16E+04	2.42E+03	5.41E-05	4.01E-03	2.42E+03	9.16E+04
Ag-110m	4.31E+02	1.14E+01	1.74E-05	1.29E-03	1.14E+01	4.31E+02
Sb-124	1.18E+02	3.11E+00	N/A	0.00E+00	3.11E+00	1.18E+02
Te-129m	7.21E+03	1.90E+02	2.50E-03	1.85E-01	1.90E+02	7.21E+03
Te-131m	2.73E+04	7.22E+02	6.30E-03	4.67E-01	7.22E+02	2.73E+04
Te-132	2.02E+05	5.35E+03	7.00E-02	5.19E+00	5.35E+03	2.02E+05
Cs-134	3.01E+04	7.96E+02	2.00E+00	1.48E+02	7.94E+02	3.00E+04
Cs-136	1.00E+04	2.65E+02	1.00E+00	7.42E+01	2.64E+02	9.95E+03
Cs-137	2.45E+04	6.47E+02	1.20E+00	8.90E+01	6.46E+02	2.44E+04
Cs-138	2.71E+05	7.17E+03	3.70E-01	2.75E+01	7.17E+03	2.71E+05
Ba-140	2.50E+05	6.61E+03	3.90E-03	2.89E-01	6.61E+03	2.50E+05
La-140	2.58E+05	6.81E+03	3.90E-03	2.89E-01	6.81E+03	2.58E+05
Ce-141	2.36E+05	6.24E+03	4.20E-04	3.12E-02	6.24E+03	2.36E+05
Ce-144	2.02E+05	5.35E+03	1.20E-04	8.90E-03	5.35E+03	2.02E+05
Pr-143	2.15E+05	5.68E+03	8.80E-05	6.53E-03	5.68E+03	2.15E+05
Np-239	2.72E+06	7.19E+04	2.04E-02	1.51E+00	7.19E+04	2.72E+06

Attachment 4
Basis of Source Term for Accident Liquid Radioactive Effluent Release

Basis for Surrogate Plan Representing Vendor Provided Information

The activity concentration values selected for the CRN PPE represent the highest values provided by the SMR vendors. In addition, the values selected for the CRN PPE are conservative when compared to the conservative composite table developed for PSEG PPE. Therefore, the accident liquid source term values selected for the CRN PPE are shown to be not unreasonable.

References

1. PSEG Site ESP Application Part 2, Site Safety Analysis Report, Chapter 1, ADAMS number ML15169A281

Attachment 5
Basis of Source Term for Accident Gaseous Radioactive Effluent Release

Guidance from Nuclear Energy Institute (NEI) 10-01 was used to determine the accident gaseous release source terms. The definition of the parameter noted in NEI 10-01 is "The activity, by radionuclide, contained in post-accident airborne effluents." The release activities are used in the Site Safety Analysis Report (Chapter 15, Accident and Transient Analysis) to evaluate the doses to the public during a postulated accident at the plant and to ensure 10 CFR 52.17 dose limits will be met.

In accordance with the guidelines of NEI 10-01, source term information was obtained for the accident gaseous effluent release activities. The small modular reactor (SMR) vendors provided the information with various levels of computational maturity. Additionally, the vendors provided source terms resulting from a loss of coolant accident (LOCA) scenario. The vendor provided source term information associated with LOCAs is presented in Table 9.

The source terms provided by one of the vendors were determined to be the most limiting. The source terms are utilized by the vendors to develop dose consequences for each SMR plant and are generally proportional to the core power. The dose consequences provided from the vendor with the limiting source terms were also bounding for the SMR vendors. Subsequently, the values from this vendor were selected as the surrogate plant values.

Some of the baseline assumptions utilized to determine these values are:

- Core melt is based on Regulatory Guide 1.183 methodology with containment leakage reduction after 24 hours
- Additional containment removal processes as licensed previously
- A decontamination factor was used for the outside containment pool water

Design certification of the SMR core design and analysis have not been completed and it is not practical to determine bounding computations until the combined license application (COLA) submittal, when a specific design has been selected. However, key input parameters have been evaluated to assess reasonableness of the dose evaluation.

The source terms considered are based on designs that utilize standard light water reactor (LWR) fuel with power levels below 800 MW thermal. Similar methodologies and analytical techniques will be utilized in computation of source terms during the vendor design certification process. As such, a comparison of the surrogate plant LOCA source terms to those of the previously approved AP1000 designs is performed. It is noted that the Vogtle Early Site Permit Application (ESPA) considered activity releases from a spectrum of postulated piping breaks in the presentation of their source terms. The CRN vendor evaluations only consider the single largest break size for Clinch River Nuclear (CRN) Site. This analysis method incorporates inherent conservatisms over the CRN source terms.

By consideration of ratio of the power levels between the CRN surrogate plant and those presented in the Vogtle ESPA (Reference 1), an estimate of the CRN source terms can be approximated. Based on a reduction of the Vogtle LOCA total activity by a factor of 0.235 (800MWt / 3400MWt) gives a total activity of 8.3E+3 Ci (3.53E+4 x 0.235) for the worst 2-hour time period. This value is approximately 25 percent larger than the CRN surrogate plant total activity for the same time period (6.64E+3) and may indicate an under prediction of vendor determined source terms.

However, any under prediction of vendor determined source terms is compensated by the margin between the predicted dose at the exclusion area boundary (EAB) and the acceptance limit and the margin in the methodology used to determine the CRN site specific atmospheric dispersion values at the EAB.

Based on the NEI 10-01 guidance document methodology, the predicted dose at the EAB (21.6 rem TEDE) is below the 10 CFR 52.17 acceptance limit by approximately 15%.

Site Dose = (21.6 rem TEDE x 1.15) = 24.84 rem TEDE (~25)

An additional margin of ~50% is included in the methodology utilized to determine the CRN site atmospheric dispersion factors. The dispersion factors utilized in the computation are based on the analytical EAB of 1,100 ft from the release boundary. However, this value contains inherent margin due to the fact that the actual EAB is in excess of 1,400 ft from the release boundary. The additional distance results in a significant improvement in the dispersion factors that are imbedded in the conservative methodology. This has a proportional resultant effect on the computed EAB dose value.

X/Q (1,100 feet) = 4.96E-3 (CRN ESPA Site Safety Analysis Report [SSAR] Table 2.3.4-12)

X/Q (1,400 feet) = 3.3E-3 (CRN ESPA SSAR Figure 1.2-2 - scaled)

subsequently, 3.3E-3 x 1.5 = 4.95E-3 (~4.96E-3)

The additional margin in these two parameters provides assurance that the source terms are not unreasonable and would likely bound any source term under prediction for application to doses at the CRN EAB.

Similarly, the Vogtle ESPA loss of coolant accident (LOCA) total activity at the low population zone (LPZ) was reduced from 8.79E+5 Ci to 2.0E+5 Ci based on ratio of operating core power levels for assessment of impacts at the LPZ. Although the reduced Vogtle value is approximately 85% higher than the CRN value, doses predicted at the LPZ for CRN are approximately a factor of 8 less than the 10 CFR 52.17 acceptance limit. Again, this margin is sufficient to predict that the offsite doses at the CRN LPZ will likely not exceed 10 CFR 52.17 limits and the source terms are not unreasonable for the CRN ESPA.

TVA has concluded that the source terms considered for the surrogate plant are representative of the potential SMR designs. The key parameter that supports this conclusion is core power. The surrogate plant considers a core power that is substantially larger (greater than 30%) than the remaining potential SMR designs. Although it is recognized that core power does not necessarily result in a linear ratio of activity release, similar methodologies and analytical techniques were utilized by all vendors in computation of offsite activity releases and will result in offsite doses that are representative of the surrogate plant. Additionally, the core burnup considered by the surrogate plant is 51 GWD/MTU; the core burnup considered by subsequent SMR designers is projected to be less than 41 GWD/MTU. Subsequently, the margin created by the core power and core burnup assumed by the surrogate plant will compensate for any variances in specific designs by other SMR vendors.

Attachment 5
Basis of Source Term for Accident Gaseous Radioactive Effluent Release

Supporting Information

Vogtle ESPA LOCA Source Term (worst 2 hours) = 3.53E+4 Ci (Reference 1)

Vogtle ESPA LOCA Source Term (worst 2 hours) ratio to CRN = 3.53E+4 Ci x (800/3400) = 8.3E+3 Ci

CRN ESPA LOCA Source Term (worst 2 hours) = 6.64E+3 Ci (Table 9)

Vogtle Ratio Value is slightly higher (25%) than CRN

CRN dose at EAB = 21.6 Rem (ESPA SSAR Table 15-1)

Margin to 10CFR52.17 Acceptance Limit = ~ 15% (21.6 x 1.15 = 25)

X/Q @ 1100 ft = 4.96E-3 (CRN ESPA SSAR Section 2.3.4)

X/Q @ 1400 ft = ~ 3.3E-3 (Reference 2)

Margin to EAB ~ 50% (3.3E-3 x 1.50 = 4.95E-3)

References

1. Vogtle Early Site Permit (ESP) Application, Rev. 5, Part 2, Site Safety Analysis Report, Chapter 15, Accident Analysis, Sections 15.1 Through 15.4, Table 15-9, 12/31/2008. NRC ML091540824.

2. Sensitivity Analysis - Atmospheric Dispersion Factors (X/Qs) at 1,400 ft from release boundary using PAVAN (results attached)

Results of Sensitivity Study Analysis - Atmospheric Dispersion Factors (X/Qs) at 1,400 ft from release boundary using PAVAN (Reference 2).

Table 1. 0.5% Max *X*/Q Values (sec/m^3) per Sector at the Exclusion Area Boundary (EAB)

0.5% Max X/Q Values (sec/m^3) per Sector at the EAB (1400 ft)						
2011-2013 Meteorology						
Plume Sector Direction	**0-2 Hours**	**0-8 Hours**	**8-24 Hours**	**1-4 Days**	**4-30 Days**	**Annual Average**
S	1.69E-03	9.06E-04	6.63E-04	3.37E-04	1.28E-04	3.89E-05
SSW	1.55E-03	8.30E-04	6.07E-04	3.08E-04	1.16E-04	3.53E-05
SW	1.65E-03	8.96E-04	6.60E-04	3.40E-04	1.31E-04	4.09E-05
WSW	2.28E-03	1.25E-03	9.25E-04	4.82E-04	1.89E-04	6.02E-05
W	2.99E-03	1.69E-03	1.28E-03	6.89E-04	2.84E-04	9.64E-05
WNW	**3.30E-03**	**1.94E-03**	**1.49E-03**	**8.35E-04**	**3.65E-04**	**1.32E-04**
NW	3.04E-03	1.76E-03	1.34E-03	7.44E-04	3.18E-04	1.13E-04
NNW	2.73E-03	1.48E-03	1.08E-03	5.56E-04	2.13E-04	6.59E-05
N	2.31E-03	1.21E-03	8.71E-04	4.31E-04	1.57E-04	4.54E-05
NNE	1.94E-03	9.84E-04	7.01E-04	3.35E-04	1.16E-04	3.19E-05
NE	1.50E-03	7.92E-04	5.76E-04	2.89E-04	1.07E-04	3.18E-05
ENE	2.11E-03	1.13E-03	8.27E-04	4.20E-04	1.58E-04	4.81E-05
E	2.16E-03	1.19E-03	8.86E-04	4.65E-04	1.85E-04	5.96E-05
ESE	2.41E-03	1.38E-03	1.04E-03	5.68E-04	2.38E-04	8.20E-05
SE	2.07E-03	1.17E-03	8.82E-04	4.76E-04	1.96E-04	6.63E-05
SSE	1.82E-03	9.77E-04	7.16E-04	3.65E-04	1.39E-04	4.26E-05

Table 2. Exclusion Area Boundary (EAB) *X*/Q Values (sec/m^3)

0.5% and 5% X/Q Values (sec/m^3) at the EAB (1400 ft)				
2011-2013 Meteorology				
Analytical EAB	**Time Period**	**Direction-Dependent *X*/Q**		**Direction Independent *X*/Q**
		0.5% Maximum	***Maximum Sector***	***5% Site Limit***
1400 ft EAB	0-2 Hours	3.30E-03	WNW	3.48E-03

Attachment 5
Basis of Source Term for Accident Gaseous Radioactive Effluent Release

Vendor 1

TABLE 9

One Unit					
Time Period / Radionuclide	0 to 2 hours	0 to 8 hours	8 to 24 hours	24 to 96 hours	96 to 720 hours
I-129	6.87E-06	1.42E-05	1.51E-06	1.50E-06	1.07E-05
I-130	2.24E-04	4.40E-04	2.61E-05	4.20E-06	5.07E-08
I-131	9.39E+01	1.94E+02	1.98E+01	1.69E+01	4.17E+01
I-132	4.83E+01	8.26E+01	1.87E+00	9.49E-01	8.44E-01
I-133	1.31E+00	2.62E+00	1.96E-01	5.92E-02	4.35E-03
I-135	6.94E-06	1.31E-05	4.84E-07	2.66E-08	8.38E-12
Kr-85	8.93E+00	6.88E+01	1.60E+02	7.17E+02	6.11E+03
Xe-131m	6.31E+00	4.83E+01	1.10E+02	4.52E+02	2.01E+03
Xe-133	5.35E+02	4.06E+03	8.84E+03	3.14E+04	6.23E+04
Xe-133m	6.01E+00	4.46E+01	8.91E+01	2.33E+02	1.46E+02
Xe-135	8.67E-03	5.41E-02	5.53E-02	2.31E-02	9.64E-05
Xe-135m	2.61E-06	9.73E-06	6.41E-06	1.47E-06	7.81E-10
Co-58	7.60E-03	1.71E-02	1.62E-03	3.24E-04	3.11E-06
Co-60	4.35E-03	9.81E-03	9.32E-04	1.88E-04	1.88E-06
Se-79	2.33E-06	5.26E-06	5.00E-07	1.01E-07	1.01E-09
Br-82	2.02E-02	4.12E-02	3.00E-03	3.93E-04	6.13E-07
Rb-86	2.08E-01	4.20E-01	3.58E-02	6.98E-03	5.94E-05
Sr-89	6.58E+00	1.48E+01	1.40E+00	2.79E-01	2.63E-03
Sr-90	1.01E+00	2.28E+00	2.16E-01	4.36E-02	4.37E-04
Sr-91	4.16E-05	8.72E-05	4.08E-06	1.87E-07	4.07E-12
Y-90	1.49E-02	5.88E-02	2.74E-02	1.37E-02	3.36E-04
Y-91	8.81E-02	1.99E-01	1.88E-02	3.74E-03	3.56E-05
Y-91m	6.90E-06	2.74E-05	2.54E-06	1.17E-07	2.57E-12
Y-93	1.38E-06	2.90E-06	1.43E-07	7.33E-09	2.49E-13
Zr-95	1.24E-01	2.80E-01	2.65E-02	5.28E-03	5.04E-05
Zr-97	1.39E-04	3.01E-04	1.91E-05	1.62E-06	4.28E-10
Nb-95	1.34E-01	3.02E-01	2.87E-02	5.79E-03	5.74E-05
Nb-95m	1.24E-03	2.78E-03	2.63E-04	5.21E-05	4.89E-07
Nb-97	1.54E-04	3.30E-04	2.06E-05	1.74E-06	4.64E-10
Nb-97m	1.36E-04	2.94E-04	1.86E-05	1.58E-06	4.17E-10
Mo-99	3.04E-01	6.78E-01	5.80E-02	9.25E-03	3.26E-05
Tc-99	2.68E-05	6.04E-05	5.74E-06	1.16E-06	1.16E-08
Tc-99m	2.92E-01	6.51E-01	5.55E-02	8.85E-03	3.13E-05
Ru-103	1.33E+00	2.99E+00	2.82E-01	5.59E-02	5.18E-04
Ru-106	5.71E-01	1.29E+00	1.22E-01	2.46E-02	2.44E-04
Rh-103m	1.33E+00	2.99E+00	2.82E-01	5.59E-02	5.18E-04
Rh-105	4.21E-02	9.30E-02	7.26E-03	9.54E-04	1.50E-06
Rh-106	5.71E-01	1.29E+00	1.22E-01	2.46E-02	2.44E-04
Pd-109	5.02E-05	1.08E-04	6.15E-06	4.24E-07	4.85E-11
Pd-112	9.46E-05	2.06E-04	1.41E-05	1.41E-06	7.24E-10
Ag-109m	4.89E-05	1.06E-04	6.15E-06	4.24E-07	4.85E-11
Ag-112	8.48E-06	5.09E-05	1.41E-05	1.65E-06	8.55E-10
Sb-124	1.46E-02	3.29E-02	3.11E-03	6.20E-04	5.90E-06
Sb-125	2.45E-01	5.51E-01	5.24E-02	1.06E-02	1.06E-04
Sb-126	6.42E-03	1.44E-02	1.34E-03	2.56E-04	2.01E-06
Sb-127	4.98E-01	1.11E+00	9.82E-02	1.68E-02	7.95E-05
Sb-128	7.34E-07	1.53E-06	6.98E-08	3.03E-09	5.23E-14
Te-125m	4.79E-02	1.08E-01	1.03E-02	2.07E-03	2.10E-05
Te-127	7.35E-01	1.65E+00	1.50E-01	2.75E-02	1.92E-04
Te-127m	2.88E-01	6.48E-01	6.15E-02	1.23E-02	1.21E-04
Te-129	5.33E-01	1.20E+00	1.13E-01	2.23E-02	2.04E-04
Te-129m	8.44E-01	1.90E+00	1.79E-01	3.54E-02	3.24E-04
Te-131	1.72E-02	3.75E-02	2.81E-03	3.42E-04	3.96E-07
Te-131m	7.58E-02	1.67E-01	1.26E-02	1.53E-03	1.77E-06
Te-132	6.13E+00	1.37E+01	1.19E+00	1.97E-01	8.09E-04
Cs-134	2.77E+01	5.62E+01	4.86E+00	9.80E-01	9.78E-03
Cs-135	1.37E-04	2.78E-04	2.41E-05	4.85E-06	4.86E-08
Cs-136	6.38E+00	1.29E+01	1.09E+00	2.09E-01	1.66E-03
Cs-137	2.27E+01	4.60E+01	3.98E+00	8.03E-01	8.05E-03
Ba-136m	9.56E-01	1.94E+00	1.64E-01	3.14E-02	2.49E-04
Ba-137m	1.99E+01	4.19E+01	3.77E+00	7.60E-01	7.62E-03
Ba-140	9.23E+00	2.08E+01	1.93E+00	3.70E-01	2.92E-03
La-140	1.73E-01	7.53E-01	3.70E-01	1.70E-01	2.87E-03
Ce-141	2.76E-01	6.22E-01	5.86E-02	1.16E-02	1.06E-04
Ce-143	8.65E-03	1.91E-02	1.47E-03	1.87E-04	2.58E-07
Ce-144	2.68E-01	6.05E-01	5.74E-02	1.16E-02	1.15E-04
Pr-142	1.02E-05	2.21E-05	1.46E-06	1.36E-07	5.27E-11
Pr-143	9.01E-02	2.03E-01	1.89E-02	3.64E-03	2.93E-05
Pr-144	1.95E-01	5.22E-01	5.74E-02	1.16E-02	1.15E-04
Pr-144m	3.45E-03	8.47E-03	8.61E-04	1.73E-04	1.72E-06
Nd-147	3.26E-02	7.32E-02	6.78E-03	1.29E-03	9.78E-06
Pm-147	1.96E-02	4.42E-02	4.20E-03	8.48E-04	8.50E-06
Pm-148	5.74E-03	1.29E-02	1.16E-03	2.08E-04	1.23E-06
Pm-148m	3.86E-03	8.70E-03	8.21E-04	1.63E-04	1.52E-06
Pm-149	4.72E-03	1.05E-02	8.75E-04	1.32E-04	3.68E-07
Pm-151	2.43E-04	5.34E-04	3.97E-05	4.72E-06	4.87E-09
Sm-151	5.72E-05	1.29E-04	1.23E-05	2.47E-06	2.48E-08
Sm-153	2.78E-03	6.17E-03	5.05E-04	7.33E-05	1.75E-07
Eu-152	1.42E-06	3.20E-06	3.04E-07	6.13E-08	6.14E-10
Eu-154	7.67E-04	1.73E-03	1.64E-04	3.31E-05	3.32E-07
Eu-155	2.95E-04	6.64E-04	6.31E-05	1.27E-05	1.27E-07
Eu-156	1.15E-02	2.58E-02	2.40E-03	4.64E-04	3.80E-06
Eu-157	7.04E-07	1.52E-06	9.17E-08	7.08E-09	1.28E-12
U-235m	4.70E-05	1.65E-04	2.22E-05	4.48E-06	4.50E-08
Np-238	7.61E-03	1.69E-02	1.40E-03	2.09E-04	5.52E-07
Np-239	4.60E-01	1.02E+00	8.60E-02	1.32E-02	3.94E-05
Pu-237	8.73E-07	1.97E-06	1.86E-07	3.69E-08	3.45E-10
Pu-238	8.23E-04	1.85E-03	1.76E-04	3.56E-05	3.57E-07
Pu-239	1.04E-04	2.34E-04	2.22E-05	4.48E-06	4.50E-08
Pu-240	1.32E-04	2.97E-04	2.82E-05	5.69E-06	5.70E-08
Pu-241	3.96E-02	8.93E-02	8.49E-03	1.71E-03	1.72E-05
Pu-242	4.60E-07	1.04E-06	9.86E-08	1.99E-08	1.99E-10
Am-241	2.29E-05	5.16E-05	4.92E-06	1.00E-06	1.04E-08
Am-242	5.50E-06	1.19E-05	7.36E-07	5.96E-08	1.32E-11
Am-243	1.99E-06	4.48E-06	4.26E-07	8.58E-08	8.60E-10
Cm-242	4.58E-03	1.03E-02	9.79E-04	1.97E-04	1.93E-06
Cm-243	1.88E-06	4.24E-06	4.03E-07	8.14E-08	8.15E-10
Cm-244	2.16E-04	4.88E-04	4.64E-05	9.35E-06	9.36E-08
Total	8.10E+02	4.73E+03	9.24E+03	3.29E+04	7.06E+04

Vendor 2

TABLE 9

Nuclide	1.4 to 3.4 hr.	0 to 8 hr.	8 to 24 hr.	24 to 96 hr.	96 to 720 hr.
I-130	8.7E+00	1.7E+01	8.3E-01	1.1E-01	2.0E-03
I-131	2.6E+02	5.4E+02	4.1E+01	3.7E+01	1.1E+02
I-132	1.9E+02	3.3E+02	2.5E+00	2.3E-03	0.0E+00
I-133	5.0E+02	1.0E+03	5.9E+01	1.6E+01	1.6E+00
I-134	1.0E+02	1.8E+02	4.6E-02	1.0E-08	0.0E+00
I-135	4.0E+02	7.6E+02	2.4E+01	9.4E-01	4.9E-04
Kr-85m	2.2E+02	5.8E+02	2.9E+02	1.3E+01	1.9E-04
Kr-85	1.3E+01	4.6E+01	1.1E+02	2.5E+02	2.1E+03
Kr-87	1.7E+02	3.0E+02	7.7E+00	6.3E-04	0.0E+00
Kr-88	4.8E+02	1.1E+03	2.6E+02	2.7E+00	6.3E-08
Xe-131m	1.3E+01	4.5E+01	1.0E+02	2.1E+02	8.6E+02
Xe-133m	6.8E+01	2.4E+02	4.9E+02	6.3E+02	4.0E+02
Xe-133	2.3E+03	8.0E+03	1.8E+04	3.2E+04	6.3E+04
Xe-135m	1.6E+00	5.5E+00	3.3E-08	0.0E+00	0.0E+00
Xe-135	4.9E+02	1.5E+03	1.6E+03	3.3E+02	1.3E+00
Xe-138	4.8E+00	1.9E+01	2.4E-08	0.0E+00	0.0E+00
Rb-86	4.7E-01	9.8E-01	4.6E-02	1.5E-02	7.9E-02
Cs-134	4.0E+01	8.3E+01	4.0E+00	1.4E+00	1.2E+01
Cs-136	1.1E+01	2.3E+01	1.1E+00	3.5E-01	1.5E+00
Cs-137	2.3E+01	4.8E+01	2.3E+00	8.2E-01	7.1E+00
Cs-138	2.3E+01	5.1E+01	3.4E-04	0.0E+00	0.0E+00
Sb-127	3.7E+00	7.4E+00	3.5E-01	8.8E-02	1.2E-01
Sb-129	7.9E+00	1.4E+01	2.3E-01	7.6E-04	7.6E-09
Te-127m	4.9E-01	9.7E-01	4.9E-02	1.7E-02	1.3E-01
Te-127	3.2E+00	5.9E+00	1.8E-01	4.2E-03	2.1E-05
Te-129m	1.7E+00	3.3E+00	1.7E-01	5.6E-02	3.6E-01
Te-129	2.9E+00	4.4E+00	4.2E-03	5.5E-09	0.0E+00
Te-131m	4.9E+00	9.6E+00	4.1E-01	5.2E-02	1.2E-02
Te-132	5.0E+01	9.9E+01	4.7E+00	1.1E+00	1.2E+00
Sr-89	1.4E+01	2.9E+01	1.4E+00	4.9E-01	3.5E+00
Sr-90	1.2E+00	2.5E+00	1.2E-01	4.4E-02	3.8E-01
Sr-91	1.5E+01	2.8E+01	8.4E-01	2.1E-02	1.1E-04
Sr-92	1.1E+01	1.7E+01	1.6E-01	8.0E-05	0.0E+00
Ba-139	8.4E+00	1.3E+01	2.3E-02	1.5E-07	0.0E+00
Ba-140	2.5E+01	5.0E+01	2.5E+00	7.9E-01	3.4E+00
Mo-99	3.3E+00	6.6E+00	3.1E-01	6.6E-02	5.8E-02
Tc-99m	2.3E+00	4.1E+00	9.3E-02	8.1E-04	2.1E-07
Ru-103	2.7E+00	5.3E+00	2.7E-01	9.2E-02	6.2E-01
Ru-105	1.3E+00	2.2E+00	3.8E-02	1.4E-04	1.8E-09
Ru-106	8.8E-01	1.8E+00	8.8E-02	3.1E-02	2.6E-01
Rh-105	1.6E+00	3.1E+00	1.4E-01	2.0E-02	6.4E-03
Ce-141	6.0E-01	1.2E+00	6.0E-02	2.0E-02	1.3E-01
Ce-143	5.3E-01	1.0E+00	4.5E-02	6.3E-03	1.8E-03
Ce-144	4.5E-01	9.1E-01	4.6E-02	1.6E-02	1.3E-01
Pu-238	1.4E-03	2.8E-03	1.4E-04	5.0E-05	4.4E-04
Pu-239	1.2E-04	2.5E-04	1.3E-05	4.4E-06	3.8E-05
Pu-240	1.8E-04	3.7E-04	1.8E-05	6.5E-06	5.6E-05
Pu-241	4.1E-02	8.2E-02	4.1E-03	1.5E-03	1.3E-02
Np-239	6.9E+00	1.4E+01	6.3E-01	1.3E-01	8.8E-02
Y-90	1.2E-02	2.5E-02	1.1E-03	2.5E-04	2.1E-04
Y-91	1.8E-01	3.7E-01	1.8E-02	6.4E-03	4.6E-02
Y-92	1.2E-01	2.1E-01	2.8E-03	4.4E-06	0.0E+00
Y-93	1.9E-01	3.5E-01	1.1E-02	3.1E-04	2.2E-06
Nb-95	2.5E-01	4.9E-01	2.5E-02	8.4E-03	5.5E-02
Zr-95	2.5E-01	4.9E-01	2.5E-02	8.5E-03	6.3E-02
Zr-97	2.2E-01	4.2E-01	1.6E-02	1.0E-03	5.7E-05
La-140	2.6E-01	5.1E-01	2.3E-02	3.6E-03	1.5E-03
La-141	1.6E-01	2.8E-01	4.2E-03	9.9E-06	3.1E-11
La-142	8.3E-02	1.3E-01	3.2E-04	5.2E-09	0.0E+00
Nd-147	9.5E-02	1.9E-01	9.4E-03	2.9E-03	1.1E-02
Pr-143	2.1E-01	4.3E-01	2.1E-02	6.8E-03	3.0E-02
Am-241	1.9E-05	3.7E-05	1.9E-06	6.6E-07	5.7E-06
Cm-242	4.4E-03	8.7E-03	4.4E-04	1.5E-04	1.2E-03
Cm-244	5.3E-04	1.1E-03	5.4E-05	1.9E-05	1.6E-04
Total	5.2E+03	1.5E+04	2.1E+04	3.3E+04	6.6E+04

Vendor 3

TABLE 9

One Unit					
Time Period / Radionuclide	0 to 2 hour	0 to 8 hours	8 to 24 hour	24 to 96 hours	96 to 720 hours
Co-58	5.98E-06	7.06E-06	--	--	--
Co-60	6.08E-05	7.18E-05	-	-	-
Kr-85	5.71E+00	3.69E+01	8.30E+01	1.87E+02	1.60E+03
Kr-85m	4.99E+01	2.09E+02	9.26E+01	4.08E+00	-
Kr-87	3.93E+01	8.16E+01	1.64E+00	-	-
Kr-88	1.05E+02	3.58E+02	7.22E+01	6.90E-01	-
Rb-86	7.20E-02	8.50E-02	-	-	-
Sr-89	9.32E-01	1.10E+00	-	-	-
Sr-90	1.51E-01	1.79E-01	-	-	-
Sr-91	1.04E+00	1.22E+00	-	-	-
Sr-92	7.17E-01	8.09E-01	-	-	-
Y-90	1.91E-03	2.66E-03	-	-	-
Y-91	2.72E-04	3.72E-04	-	-	-
Y-92	1.65E-01	2.14E-01	-	-	-
Y-93	8.45E-06	9.85E-06	-	-	-
Zr-95	9.53E-06	1.13E-05	-	-	-
Zr-97	8.57E-06	1.00E-05	-	-	-
Nb-95	1.09E+01	1.29E+01	-	-	-
Mo-99	1.07E+01	1.26E+01	-	-	-
Tc-99m	9.75E+00	1.15E+01	-	-	-
Ru-103	3.12E-01	3.69E-01	-	-	-
Ru-105	1.86E-01	2.14E-01	-	-	-
Ru-106	1.92E-01	2.27E-01	-	-	-
Rh-105	2.36E-01	2.78E-01	-	-	-
Sb-127	4.13E+00	4.87E+00	-	-	-
Sb-129	8.71E+00	9.97E+00	-	-	-
Te-127	4.09E+00	4.83E+00	-	-	-
Te-127m	2.05E-01	2.42E-01	-	-	-
Te-129	9.65E+00	1.10E+01	-	-	-
Te-129m	1.13E-02	1.37E-02	-	-	-
Te-131m	5.80E+00	6.81E+00	-	-	-
Te-132	4.98E+01	5.87E+01	-	-	-
I-131	4.28E+01	7.09E+01	5.40E+01	1.04E+02	3.13E+02
I-132	5.85E+01	8.28E+01	3.11E+00	1.10E-02	-
I-133	8.12E+01	1.30E+02	6.61E+01	4.24E+01	4.22E+00
I-134	3.10E+01	3.42E+01	1.30E-02	-	-
I-135	6.45E+01	9.55E+01	1.98E+01	2.22E+00	-
Xe-133	3.64E+02	2.31E+03	4.89E+03	8.68E+03	1.72E+04
Xe-135	1.96E+02	1.03E+03	1.01E+03	2.11E+02	9.00E-01
Cs-134	1.21E+01	1.43E+01	-	-	-
Cs-136	2.43E+00	2.87E+00	-	-	-
Cs-137	7.89E+00	9.32E+00	-	-	-
Ba-139	5.32E-01	5.80E-01	-	-	-
Ba-140	1.30E+00	1.54E+00	-	-	-
La-140	2.61E-02	3.62E-02	-	-	-
La-141	6.64E-06	7.59E-06	-	-	-
La-142	3.86E-06	4.24E-06	-	-	-
Ce-141	8.90E-06	1.50E-05	-	-	-
Ce-143	8.77E-06	1.03E-05	-	-	-
Ce-144	7.42E-06	8.76E-06	-	-	-
Pr-143	8.65E-06	1.02E-05	-	-	-
Nd-147	3.45E-06	4.08E-06	-	-	-
Np-239	1.37E-04	1.61E-04	-	-	-
Pu-238	5.89E-08	6.96E-08	-	-	-
Pu-239	4.13E-09	4.87E-09	-	-	-
Pu-240	7.25E-09	8.56E-09	-	-	-
Pu-241	1.73E-06	2.04E-06	-	-	-
Am-241	4.17E-09	4.93E-09	-	-	-
Cm-242	9.92E-07	1.17E-06	-	-	-
Cm-244	2.44E-07	2.88E-07	-	-	-
Notes:					
1. Preliminary information, currently under development.					
Total	1.18E+03	4.61E+03	6.29E+03	9.23E+03	1.91E+04

Vendor 4

TABLE 9

Activity Releases (CI)					
Nuclide	Worst 2 hour	0-8 hour	8-24 hour	1-4 days	4-30 days
Kr-85m	3.51E+02	9.28E+02	4.60E+02	2.10E+01	0.00E+00
Kr-85	3.01E+01	1.05E+02	2.50E+02	5.64E+02	4.84E+03
Kr-87	2.66E+02	4.84E+02	1.22E+01	1.00E-02	0.00E+00
Kr-88	7.48E+02	1.74E+03	4.05E+02	4.20E+00	0.00E+00
Xe-131m	1.92E+01	6.69E+01	1.55E+02	3.13E+02	1.27E+03
Xe-133m	1.17E+00	3.98E+00	8.14E+00	1.07E+01	6.72E+00
Xe-133	3.82E+03	1.32E+04	2.95E+04	5.25E+04	1.04E+05
Xe-135m	2.08E+00	8.91E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	8.53E+02	2.57E+03	2.70E+03	5.62E+02	2.30E+00
Xe-138	5.81E+00	2.92E+01	0.00E+00	0.00E+00	0.00E+00
I-130	2.12E+00	4.19E+00	1.55E-01	8.10E-03	2.00E-04
I-131	1.34E+02	2.76E+02	1.52E+01	5.80E+00	1.75E+01
I-132	9.61E+01	1.69E+02	1.11E+00	1.00E-07	0.00E+00
I-133	2.59E+02	5.20E+02	2.28E+01	2.54E+00	2.50E-01
I-134	4.98E+01	9.21E+01	2.10E-02	0.00E+00	0.00E+00
I-135	2.06E+02	3.94E+02	1.02E+01	1.50E-01	0.00E+00
Cs-134	2.35E+01	4.71E+01	2.06E+00	1.11E-02	9.60E-02
Cs-136	6.70E+00	1.20E+01	4.60E-01	3.09E-03	9.00E-03
Cs-137	1.80E+01	3.63E+01	1.59E+00	9.07E-03	7.50E-02
Cs-138	1.14E+01	2.75E+01	1.00E-08	0.00E+00	0.00E+00
Rb-86	2.06E-01	4.15E-01	1.81E-02	9.07E-05	4.80E-04
Te-127m	2.74E-01	5.48E-01	2.62E-02	1.40E-04	1.11E-03
Te-127	1.34E+00	2.52E+00	7.40E-02	0.00E+00	0.00E+00
Te-129m	9.09E-01	1.82E+00	8.65E-02	4.00E-04	3.00E-03
Te-129	1.14E+00	1.80E+00	1.70E-03	0.00E+00	0.00E+00
Te-131m	3.32E+00	6.51E+00	2.67E-01	5.00E-04	2.00E-04
Te-132	2.59E+01	5.14E+01	2.32E+00	8.00E-03	9.00E-03
Sb-127	1.59E+00	3.17E+00	1.44E-01	6.00E-04	7.00E-04
Sb-129	3.38E+00	5.99E+00	9.99E-02	0.00E+00	0.00E+00
Sr-89	7.79E+00	1.56E+01	7.42E-01	4.00E-03	2.80E-02
Sr-90	9.52E-01	1.91E+00	9.12E-02	5.00E-04	4.30E-03
Sr-91	8.01E+00	1.51E+01	4.46E-01	0.00E+00	0.00E+00
Sr-92	5.48E+00	9.27E+00	8.32E-02	0.00E+00	0.00E+00
Ba-139	4.14E+00	6.61E+00	1.17E-02	0.00E+00	0.00E+00
Ba-140	1.33E+01	2.66E+01	1.25E+00	6.00E-03	2.60E-02
Ru-103	1.40E+00	2.81E+00	1.34E-01	7.00E-04	4.80E-03
Ru-105	6.27E-01	1.12E+00	1.93E-02	0.00E+00	0.00E+00
Ru-106	4.75E-01	9.52E-01	4.55E-02	2.50E-04	2.08E-03
Rh-105	8.37E-01	1.65E+00	6.92E-02	2.00E-04	0.00E+00
Mo-99	1.71E+00	3.38E+00	1.51E-01	5.00E-04	4.00E-04
Tc-99m	1.16E+00	2.12E+00	4.78E-02	0.00E+00	0.00E+00
Ce-141	3.19E-01	6.38E-01	3.03E-02	1.60E-04	1.02E-03
Ce-143	2.81E-01	5.53E-01	2.30E-02	5.00E-05	2.00E-05
Ce-144	2.65E-01	5.31E-01	2.54E-02	1.40E-04	1.15E-03
Pu-238	5.94E-04	1.19E-03	5.68E-05	3.00E-07	2.70E-06
Pu-239	7.10E-05	1.42E-04	6.79E-06	4.00E-08	3.20E-07
Pu-240	1.08E-04	2.16E-04	1.03E-05	6.00E-08	4.80E-07
Pu-241	2.64E-04	5.30E-04	2.53E-05	1.40E-07	1.19E-06
Np-239	3.16E+00	6.26E+00	2.76E-01	8.00E-04	6.00E-04
Y-90	9.55E-03	1.89E-02	8.43E-04	2.00E-06	3.00E-06
Y-91	1.01E-01	2.02E-01	9.62E-03	5.00E-05	3.80E-04
Y-92	6.37E-02	1.11E-01	1.47E-03	0.00E+00	0.00E+00
Y-93	9.72E-02	1.84E-01	5.60E-03	0.00E+00	0.00E+00
Nb-95	1.34E-01	2.69E-01	1.28E-02	7.00E-05	4.40E-04
Zr-95	1.32E-01	2.65E-01	1.26E-02	7.00E-05	5.00E-04
zr-97	1.17E-01	2.25E-01	8.22E-03	1.00E-05	0.00E+00
La-140	1.32E-01	2.61E-01	1.11E-02	3.00E-05	1.00E-05
La-142	4.13E-02	6.62E-02	1.66E-04	0.00E+00	0.00E+00
Nd-147	4.89E-02	9.77E-02	4.60E-03	2.00E-05	8.00E-05
Pr-143	1.17E-01	2.34E-01	1.11E-02	5.00E-05	2.40E-04
Am-241	1.56E-05	3.12E-05	1.49E-06	8.00E-09	7.00E-08
Cm-242	3.16E-03	6.33E-03	3.02E-04	1.70E-06	1.32E-05
Cm-244	1.84E-04	3.69E-04	1.76E-05	1.00E-07	8.30E-07
Total	6.64E+03	2.00E+04	3.31E+04	5.39E+04	1.10E+05