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

CNL-18-046

March 30, 2018

10 CFR 52.17
10 CFR 2.390

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

Clinch River Nuclear Site
NRC Docket No. 52-047

Subject: Response to Portion of Request for Additional Information Related to Emergency Planning Exemption Requests 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. USNRC Request for Additional Information No. 7, eRAI 8885, ESPA Application Section: Part 6 - Exemptions and Departures, EP Exemptions, dated July 28, 2017
3. Letter from TVA to NRC, CNL-17-101, "Response to Request for Additional Information Related to Emergency Planning Exemption Requests in Support of Early Site Permit Application for Clinch River Nuclear Site," dated August 24, 2017
4. USNRC Request for Additional Information No. 10, eRAI 9206, ESPA Application Section: Part 6 - Exemptions and Departures (Supplemental Questions to eRAI 8885), dated November 9, 2017
5. USNRC Audit Plan, "Audit of Clinch River Nuclear Site Early Site Permit Application - Part 6 - Exemptions and Departures, Emergency Planning Exemptions," dated November 15, 2017
6. Letter from TVA to NRC, CNL-18-020, "Response to Portion of Request for Additional Information Related to Emergency Planning Exemption Requests in Support of Early Site Permit Application for Clinch River Nuclear Site," dated March 9, 2018

By letter dated May 12, 2016 (Reference 1), Tennessee Valley Authority (TVA) submitted an early site permit application (ESPA) for the Clinch River Nuclear (CRN) Site in Oak Ridge, TN. Based on the staff's review of ESPA Part 6, *Exemptions and Departures*, an electronic request for additional information (eRAI) 8885 was issued (Reference 2). By letter dated August 24, 2017 (Reference 3), TVA provided a response to eRAI 8885. Based on the information provided in Reference 3, a follow-up eRAI (9206) was issued (Reference 4).

Additionally, the NRC staff identified a need for an audit related to the proposed exemptions to emergency preparedness requirements in support of the CRN Site ESPA (Reference 5). A regulatory audit was conducted from November 15, 2017 through February 9, 2018. By letter dated March 9, 2018 (Reference 6), TVA provided a response to a portion of eRAI 9206.

The purpose of this letter is to provide the TVA response to the remaining portion of eRAI 9206 and to provide supplemental information requested during the audit. Enclosure 1 provides the response to Key Issue 1: Question 1 of eRAI 9206. The response is informed by the discussions and information shared with the staff over the course of the audit. Enclosure 2 provides supplemental information related to the example analyses conducted using the NuScale Power, LLC (NuScale) design summarized in Reference 3. NuScale considers information in Enclosure 2 to this letter to be proprietary and therefore, exempt from public disclosure pursuant to 10 CFR 2.390, *Public Inspections, Requests for Withholding*. An affidavit for withholding information, executed by NuScale, is provided in Enclosure 4. Therefore, on behalf of NuScale, TVA requests Enclosure 2 be withheld from public disclosure in accordance with the associated NuScale affidavit and the provisions of 10 CFR 2.390. Enclosure 3 provides the nonproprietary version of the information in Enclosure 2.

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 30th day of March 2018.

Respectfully,

J. W. Shea
Digitally signed by J. W. Shea
DN: cn=J. W. Shea, o=Tennessee Valley Authority, ou=Nuclear Licensing, email=jwshea@tva.gov, c=US
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J. W. Shea
Vice President, Nuclear Regulatory Affairs and Support Services

Enclosures

cc: See Page 3

Enclosures:

1. TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA
2. Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA (Proprietary Version)
3. Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA (Nonproprietary Version)
4. NuScale Power, LLC Affidavit (AF-0318-59303)

cc (w/ Enclosures):

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Enclosure 1 to Letter CNL-18-046

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

ENCLOSURE 1

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

NRC Introduction

Supplemental Question to eRAI-8885

By letter dated August 24, 2017 (ADAMS Accession No. ML17237A175), the Clinch River Nuclear [CRN] *site early site permit application (ESPA) applicant, Tennessee Valley Authority (TVA) submitted a response to Request for Information (RAI) Letter No. 7, eRAI-8885. To address eRAI-8885 Question 2, TVA described a representative analysis that was done to show that the technical basis criteria for the plume exposure pathway emergency planning zone size given within Site Safety Analysis Report (SSAR) Section 13.3.3 can be met for one design included within the ESPA plant parameter envelope (PPE). The plant-related information submitted within this analysis was for the NuScale design only.*

As described in SSAR 13.3.3.1.1 "Environmental Protection Agency Protective Action Guides," the category of more frequent less severe core melt accidents includes intact containment, beyond design basis accident scenarios and accident scenarios with a mean core damage frequency (CDF) > 1×10^{-6} per reactor-year. For the less severe core melt accident category, the analysis discussed in the RAI response evaluated the dose consequences at the site boundary for the most probable scenario chosen from the internal events, at power, intact containment severe accident scenarios used to develop the NuScale design basis source term for the maximum hypothetical accident in NuScale design certification application Final Safety Analysis Report (FSAR) 15.0.3.9, which is currently under staff review.

As described in SSAR 13.3.3.1.2, "Substantial Reduction in Early Health Effects," the category of less frequent more severe core melt accidents include postulated containment failure or bypass events with mean CDF > 1×10^{-7} per reactor-year. Accident sequences with mean CDF > 1×10^{-8} per reactor-year should be considered in the initial sequence selection. The RAI response stated that there are no credible events for the NuScale design within the less frequent more severe accidents category.

Key Issue 1: TVA is using the PPE approach for the ESPA. Moreover, the analysis-related information provided in the RAI response is only specific to the NuScale design which is not the design that could potentially have the largest post-accident offsite dose consequences.

NRC RAI Key Issue 1, Question 1

1. *Please explain how providing information about one design that may fit in the Clinch River Nuclear site ESPA PPE is sufficient to support the exemption requests to the EPZ size for any plant design that may be covered by the PPE.*

ENCLOSURE 1

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

TVA Response

In the response to eRAI 8885 (Reference 1), TVA provided results of example analyses conducted using the NuScale design to demonstrate that the proposed accident consequence technical criteria described in the CRN ESPA SSAR Subsection 13.3.3 for plume exposure pathway (PEP) emergency planning zone (EPZ) can be met. The example analyses using the NuScale design was CRN Site specific and demonstrated that at least one design in the PPE could meet the technical criteria set forth in the ESPA. The NuScale design was selected for the example analyses because of the availability of substantially more detailed design and accident analysis information compared to other designs that informed the PPE. Moreover, the basis for the emergency planning (EP) exemption requests is the merit of the technical criteria described in SSAR Subsection 13.3.3 for PEP EPZ sizing and the special circumstances warranted by the unique small modular reactor (SMR) designs and not the acceptability of any one design to meet the criteria. The design features common to the SMRs that informed the PPE and the special circumstances these design features provide are discussed below.

10 CFR 50.12, *Specific Exemptions*, states that the Commission will not consider granting an exemption unless special circumstances are present. Part 6 of the ESPA discusses the special circumstances that exist at the CRN Site due to the enhanced safety features of the SMR designs under consideration. These SMR safety features and the technical basis provided in the ESPA SSAR 13.3 for PEP EPZ sizing enable meeting the underlying purpose of the regulations mentioned in the EP exemption requests. Provided below is an overview of the SMR special circumstances that justify an innovative emergency preparedness approach for SMRs at the CRN Site.

<u>*Special Circumstance # 1: Reduced Likelihood of Accidents*</u>

The reduced likelihood of accidents is demonstrated by the reduced core damage frequency (CDF) and large release frequency (LRF) values for SMRs compared to large light water reactors (LWRs). SMRs can be expected to reduce CDF values from traditional large LWRs by three orders of magnitude or more. Table 1 of this enclosure provides a comparison of CDF and LRF between a range of SMRs, traditional large LWRs, and an AP1000 reactor.

CDF and LRF reductions are supported, in part, by eliminating multiple historically considered design-basis events (DBEs). The elimination of large break loss of coolant accidents (LOCAs) is a primary example. Since there is no large-bore reactor coolant system piping, large break LOCAs are eliminated. An additional example is the elimination of events related to a loss, or reduction, of forced reactor coolant flow. By removing reactor coolant pumps and relying on natural circulation for core cooling, events related to a loss, or reduction, of forced reactor coolant flow and pump seal failures are eliminated.

Another key to reducing CDF and LRF is the reduced complexity of systems and inclusion of passive processes in those systems. Fewer safety systems with fewer components reduce opportunities for failures. SMR designs achieve safety goals with less than half the number of safety related systems compared to a traditional large LWR. Additionally, many of these systems include passive processes and eliminate opportunities for component failures and operator errors that contribute to CDF. The reduction in the number of components and systems in the SMR designs has a positive influence on the CDF and LRF values and results in designs that are influenced by Probabilistic Risk Assessment (PRA) and are significantly safer than current industry plants. Table 2 of this enclosure includes a comparison of some design parameters that reduce the likelihood of accidents for a range of SMRs, traditional large LWRs, and an AP1000 reactor.

Special Circumstance # 2: Slower Accident Progression

Slower accident progression is demonstrated by the time it takes to reach core uncovery after initiation of an event. For large LWRs, core uncovery can occur within seconds during a design basis event (DBE). For SMRs it is expected that there will be more than 96 hours until core uncovery and some designs may never uncover the core during DBEs. It can take more than 10 hours to reach core uncovery for SMR beyond design basis events (BDBEs) with CDFs greater than 1.00E-11. Table 3 of this enclosure provides a comparison of DBE accident progression parameters for a range of SMRs, traditional large LWRs, and an AP1000 reactor.

A key to slowing accident progression is the amount of coolant water available for core cooling. The more coolant water that is available compared to the heat generated by the core, the longer it will take to reach core uncovery. Primary system liquid mass to core power ratios for SMRs are expected to be more than 4 times that of a typical large LWR. Table 2 of this enclosure includes a comparison of primary coolant water available to the core for a range of SMRs, traditional large LWRs, and an AP1000 reactor.

Special Circumstance # 3: Reduced Accident Consequences

Reduced accident consequences are demonstrated by reduced doses from design basis accidents. Table 4 of this enclosure provides a comparison of LOCA design basis accident doses determined from Final Safety Analysis Reports (FSARs) for a range of SMRs, traditional large LWRs, and an AP1000 reactor. Doses provided in Table 4 of this enclosure are calculated at each design’s respective assumed Exclusion Area Boundary (EAB) distances and meteorological conditions. Considering that the SMR designs have assumed smaller EAB distances than the traditional large LWRs, the differences in dose demonstrated in Table 4 are expected to be larger when applied to similar EAB distances and meteorological conditions.

ENCLOSURE 1

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

The primary factor in reducing the dose consequences for SMRs is the reduction in source terms. Source term reductions for SMRs are primarily driven by reduced core power (i.e., less fuel in the core). Since there is less fissile material, and therefore fewer fission products and activated material, there is less radioactive material that can be released from the core. Table 2 below includes a comparison of core parameters for a range of SMRs, traditional LLWRs, and an AP1000 reactor.

Additionally, in the event of a core release, SMR designs provide for enhanced removal of radioisotopes. For example, aerosol scrubbing in submerged SMRs is improved compared to large LWRs due to the higher surface area to volume ratios. The increased deposition surface area, condensation surface area, and higher condensation rates lead to higher decontamination factors.

The SMR special circumstances described above are supported by common SMR attributes and design features that the nuclear industry recognizes in multiple SMR specific position papers (References 2, 3, and 4). Common design features of SMRs include smaller core sizes, smaller source terms, integral vessel and coolant system layouts, large coolant volume to power ratios, lower linear power density, passive heat removal systems, large containment surface area to volume ratios, submerged containments, and high pressure containments. These common design features reduce the likelihood of accidents, slow accident progression, and reduce accident consequences. Table 1 of this enclosure provides a comparison for a range of SMRs, traditional large LWRs, and an AP1000 reactor. Various design features and parameters are provided in Table 2 with descriptions of how the SMR values support the SMR special circumstances.

Table 1 - Comparison of PRA Parameters Between SMRs and LWRs

	CRN Smallest Core	CRN Largest Core	Large PWR (AP1000)	Large 4 Loop PWR (SQN)	Medium 3 Loop PWR (Turkey Point)	REMARKS
Core Damage Frequency (CDF) Internal Events	3.0E-10	<5.00E-08	2.41E-07	1.562E-05	~6.25E-05	Reduced CDF and LRF demonstrate the overall reduced likelihood of accidents for SMRs.
Large Release Frequency (LRF) Internal Events	2.1E-11	DNA	1.95E-08	2.609E-06	<1.00E-06	

DNA - Data Not Available
PWR - Pressurized Water Reactor
SQN - Sequoyah

Table 2 - Comparison of Key Design Features/Parameters Between SMRs and Large LWRs

Parameter/Feature	CRN Smallest Core	CRN Largest Core	Large PWR (AP1000)	Large 4 Loop PWR (SQN)	Medium 3 Loop PWR (Turkey Point)	REMARKS
Core Power [MWt]	160	800	3,400	3,455	2644	Lower power levels, smaller cores, and less fuel result in smaller radionuclide inventory available for release in an accident and reduced accident consequences. Additionally, the lower power levels, smaller cores, and smaller heat rates slows accident progression.
Fuel Weight [lbs UO_2]	20,350	DNA	211,588	220,213	176,000	
Number of Fuel Rods	9,768	23,496	41,448	50,952	32,028	
Active Fuel Height [Inches]	78.7	~96	168	~144	~144	
Average Linear heat rate [kilowatt/Feet (kw/ft)]	2.5	DNA	5.7	5.5	~6.7	
Source Term Total [Curies (Approximate)]	4E+04	2E+05	9E+05	1E+09[1]	8E+08[2]	
Primary Liquid Mass [lbm]	117,000	DNA	422,560	538,640	389,604	Higher primary coolant mass/power ratio reduces accidents consequences, slows accident progression, and reduces the likelihood of core damage accidents.
Primary Liquid Mass/Power Ratio [lbm/MW]	731	DNA	124	156	~147	
Containment Design Pressure [psig]	1,000 psia	DNA	59	12*	55	Higher containment design pressures hardens the third fission product barrier reducing accident consequences and the likelihood of accidents involving containment failure. (* ice condenser containment)

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 2 - Comparison of Key Design Features/Parameters Between SMRs and Large LWRs

Parameter/Feature	CRN Smallest Core	CRN Largest Core	Large PWR (AP1000)	Large 4 Loop PWR (SQN)	Medium 3 Loop PWR (Turkey Point)	REMARKS
Integrated Reactor Vessel, Steam Generator, Pressurizer	Yes	Yes	No	No	No	An integral reactor vessel hardens the second fission product barrier reducing accident consequences and the likelihood of accidents involving failure of the reactor coolant pressure boundary.
Primary Coolant External Loop Piping	None	None	2 hot 4 cold	4 loops	3 loops	

PWR - Pressurized Water Reactor
SQN - Sequoyah
DNA - Data Not Available
(1) LOCA Source Term Activities for Control room and Offsite Dose Analyses
(2) Core Source Term adjusted per Table 2 of RG 1.183 Alternate Source Term, July 2000.

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 3 - Comparison of Design Basis Accident Progression Between SMRs and Large LWRs

	CRN Smallest Core	CRN Largest Core	Large PWR (AP1000)	Large PWR (SQN)	Medium 3 Loop PWR (Turkey Point)	REMARKS
Time to initial core uncovery [sec]	None (See Note 1)	DNA	~1	~1	~5	Increased time to core uncovery and decreased time to core reflood illustrates slower accident progression and reduced likelihood of core uncovery accidents.
Core recovery reflood begins [sec]	Not Applicable (See Note 1)	DNA	54	48.2	~35	
10CFR50.46 Requirements						
Peak Cladding Temperature (PCT) [°F] [Criteria ≤2200°F]	~660 (See Note 2)	DNA	1837	<2112	2152	Post LOCA. Decreased PCT and clad oxidation illustrates slower accident progression and reduced likelihood of core damage accidents.
Maximum local clad oxidation [%] [Criteria ≤17%]	None due to no core uncovery	DNA	2.25	<6	10.46	
Maximum core-wide clad oxidation [%] [Criteria ≤1%]	None due to no core uncovery	DNA	0.2	<0.8	0.4	
Long Term cooling	Retained	DNA	Retained	Retained	Retained	

DNA - Data Not Available

PWR - Pressurized Water Reactor
SQN - Sequoyah

Note 1: Minimum level remains a few inches above the Top of Active Fuel (TAF) and returns to about 10 ft. above TAF.

Note 2: 100% RCS Injection line break. Maximum PCT occurs at 100% reactor power for this natural circulation reactor.

Table 4 - Comparison of Accident Consequences Between SMRs and Large LWRs

FSAR Design Basis LOCA	CRN Smallest Core	CRN Largest Core	Large PWR (AP1000)	Large PWR (SQN)	Medium 3 Loop PWR (Turkey Point)	REMARKS
Dose at Exclusion Area Boundary (EAB)* TEDE [REM]**	0.22	4.35	24.6	9.99	4.98	Reduced dose demonstrate the overall reduced accident consequences for SMRs.

PWR - Pressurized Water Reactor
SQN - Sequoyah
REM - roentgen equivalent man
*Doses provided are calculated at each designs respective assumed EAB distances and meteorological conditions.
**Regulatory Limit is 25 REM.

ENCLOSURE 1

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

Non-Design Specific Plant Parameter for EPZ Exemption Requests

Discussions with NRC staff have indicated value in establishing a plant parameter specific for EPZ exemption requests that will ensure the appropriate application of the exemption requests to support a Site Boundary PEP EPZ at CRN Site. This specific plant parameter is separate from the example analysis developed using the NuScale design described in Enclosure 2 of this submittal and is applicable only to Part 6 of the ESPA and will provide additional assurance that the selected reactor design will meet a Site Boundary PEP EPZ at the CRN Site. The need to develop a non-design-specific plant parameter was identified, because the ESPA is not design specific. To this end, TVA has selected atmospheric release source term as the appropriate non-design-specific plant parameter. Because the PEP EPZ sizing methodology described in SSAR Subsection 13.3.3 is based on Environmental Protective Agency (EPA) Protective Action Guides (PAGs), which evaluate a 4-day (96-hour) dose, the atmospheric release source term selected for the non-design-specific plant parameter is a total 4-day release.

To develop a non-design-specific 4-day total atmospheric release source term, TVA first created a composite source term from a spectrum of different types of accidents (of varying severity and speed) and SMR vendors. The composite source term was informed by the ESPA PPE Chapter 15 source term and two cases from NuScale's Design Basis Source Term (DBST). The NuScale DBST cases are provided in Table 2 of Enclosure 2 of this submittal. Two different cases from NuScale's DBST were included to account for varying severity and speed of accidents. The ESPA PPE Chapter 15 source term was included to account for SMR technology differences. The composite source term was created using the maximum isotopic activity from either the ESPA PPE Chapter 15 source terms or either of the two different NuScale DBST cases for each of the three major time steps, i.e., 0-8 hours, 8-24 hours, 24-96 hours. Certain isotopes included in the ESPA PPE Chapter 15 source term, such as Xe-131m, Xe-133m, Xe-138, Cs-138, and I-130, were excluded from the composite source term because they are not included in the industry standard set of 60 isotopes provided in Table 1.4.3.2-2 of NUREG/CR-6604, *RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation.* This table in NUREG/CR-6604 provides normalized MACCS sample pressurized water reactor (PWR) core inventory that can be used to compute radiological release consequences. An additional 25% of discretionary margin was added to the composite source terms for each time period to account for the design uncertainty and analysis maturity of all SMR designs. The composite source terms of each time period with the additional 25% of discretionary margin were then summed to create a composite 4-day total atmospheric release source term. This composite 4-day atmospheric release source term was then evaluated for compliance with the EPA PAG dose limits using MACCS. However, MACCS inputs are core inventory release fractions by elemental class, not isotopic activity, and MACCS calculates the decay of individual nuclides prior to release. Therefore, iterations of analysis were conducted to ensure that the atmospheric release source terms evaluated in MACCS appropriately represent the composite 4-day total atmospheric release source term developed. The 4-day total atmospheric release source term used in MACCS is the EPZ PPE source term. Table 5 of this enclosure provides the EPZ PPE source term values produced using this methodology. This methodology establishes a source term that represents a spectrum of SMR designs and applies adequate conservatism for design uncertainties. Compliance with the EPA PAG dose limits for mean and 95th percentile total effective dose equivalent (TEDE) of 1 roentgen equivalent man (rem) and 5 rem, respectively, was confirmed using the EPZ PPE source term.

Table 5 - EPZ PPE Source Term

Nuclide	4-Day Total
Kr-85	3.29E+03
Kr-85m	1.94E+03
Kr-87	1.10E+03
Kr-88	3.04E+03
Xe-133	1.74E+05
Xe-135	1.49E+04
Xe-135m	6.95E+02
Cs-134	1.26E+02
Cs-136	2.82E+01
Cs-137	8.88E+01
Rb-86	9.92E-01
Rb-88	2.59E+03
Ba-139	1.22E+01
Ba-140	4.82E+01
Sr-89	2.20E+01
Sr-90	7.46E+00
Sr-91	2.05E+01
Sr-92	1.27E+01
Ba-137m	8.00E+01
I-131	6.79E+02
I-132	4.35E+02
I-133	9.72E+02
I-134	2.08E+02
I-135	6.59E+02
Sb-127	1.51E+01
Sb-129	1.23E+01
Te-127	1.60E+01
Te-127m	2.86E+00
Te-129	1.75E+01
Te-129m	8.15E+00
Te-131m	2.22E+01
Te-132	1.78E+02
Te-131	1.09E+01
Rh-105	2.90E+00
Ru-103	4.13E+00

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TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 5 - EPZ PPE Source Term

Nuclide	4-Day Total
Ru-105	1.55E+00
Ru-106	2.68E+00
Rh-103m	4.11E+00
Rh-106	2.70E+00
Nb-95	6.45E+01
Co-58	7.88E-05
Co-60	8.74E-04
Mo-99	6.16E+01
Tc-99m	5.80E+01
Nb-97	3.95E+00
Nb-97m	4.61E-01
Ce-141	1.31E+00
Ce-143	1.09E+00
Ce-144	1.10E+00
Np-239	1.10E+01
Pu-238	7.75E-03
Pu-239	3.21E-04
Pu-240	6.48E-04
Pu-241	1.60E-01
Zr-95	6.34E-01
Zr-97	5.64E-01
Am-241	1.06E-04
Cm-242	2.61E-02
Cm-244	1.09E-02
La-140	4.75E+00
La-141	2.45E-02
La-142	8.65E-01
Nd-147	6.82E+00
Pr-143	3.10E-01
Y-90	5.05E-01
Y-91	2.74E-01
Y-92	7.46E+00
Y-93	2.90E-01
Y-91m	9.90E+00
Pr-144	9.65E-01
Pr-144m	1.72E-02

ENCLOSURE 1

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 1, Question 1, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

TVA anticipates the selected reactor design will be bounded by this newly developed EPZ PPE source term with any exceedances reviewed for acceptability at the Combined License (COL) Application stage consistent with SSAR Section 2.0, *Plant Parameter Envelope*. The applicability of the EP exemption requests will be based on the selected design adequately demonstrating conformance with the technical criteria for PEP EPZ sizing set forth in ESPA SSAR Subsection 13.3.3 and meeting the EPA PAG dose limit and the EPZ PPE source terms.

References

1. Letter from TVA to NRC, CNL-17-101, "Response to Request for Additional Information Related to Emergency Planning Exemption Requests in Support of Early Site Permit Application for Clinch River Nuclear Site," dated August 24, 2017

2. Electric Power Research Institute, Planning Guide, " Advanced Nuclear Technology: Advanced Light Water Reactor Utility Requirements Document, " December 2014

3. Nuclear Energy Institute, Position Paper, "Small Modular Reactor Source Terms," December 2012

4. International Atomic Energy Agency, "Design Safety Considerations for Water Cooled Small Modular Reactors Incorporating Lessons Learned from the Fukushima Daiichi Accident," IAEA-TECDOC-1785, 2016

Enclosure 2 to Letter CNL-18-046

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

Enclosure 3 to Letter CNL-18-046

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

ENCLOSURE 3

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

By letter dated May 12, 2016 (Reference 1), Tennessee Valley Authority (TVA) submitted an early site permit application (ESPA) for the Clinch River Nuclear (CRN) Site in Oak Ridge, TN. Based on the staff's review of ESPA Part 6, *Exemptions and Departures*, an electronic request for additional information (eRAI) 8885 was issued (Reference 2). By letter dated August 24, 2017 (Reference 3), TVA provided a response to eRAI 8885. Based on the information provided in Reference 3, a follow-up eRAI (9206) was issued (Reference 4). Additionally, the NRC staff identified a need for an audit related to the proposed exemptions to emergency preparedness requirements in support of the CRN Site ESPA (Reference 5). A regulatory audit was conducted from November 15, 2017 through February 9, 2018.

This enclosure provides the supplemental information requested during the audit in support of the staff's review of the exemption requests in Part 6 of the ESPA.

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. USNRC Request for Additional Information No. 7, eRAI 8885, ESPA Application Section: Part 6 - Exemptions and Departures, EP Exemptions, dated July 28, 2017
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5. USNRC Audit Plan, "Audit of Clinch River Nuclear Site Early Site Permit Application - Part 6 - Exemptions and Departures, Emergency Planning Exemptions," dated November 15, 2017

Supplemental Information

TVA is providing the following supplemental information associated with the example analyses conducted using the NuScale Power, LLC (NuScale) design to demonstrate that the proposed accident consequence technical criteria described in the CRN Site ESPA Site Safety Analysis Report Subsection 13.3.3 for plume exposure pathway (PEP) emergency planning zone (EPZ) can be met. The example analyses summarized in Reference 3 was updated and the results of the revised example analyses are presented below. Table 1 of this enclosure provides the updated CRN Site Boundary PEP EPZ 4-day comparison to Environmental Protection Agency (EPA) Protective Action Guide (PAG) limits. The analyses demonstrate that the mean total effective dose equivalent (TEDE) dose is bounding of the 50th percentile dose. Table 2 of this enclosure provides the total activity released to the environment over several intervals for the criterion a and b accidents analyzed in the updated example analyses.

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 1 - Updated CRN Site Boundary PEP EPZ 4-Day Dose Comparison to EPA PAG Dose Limits Using NuScale Design

Criterion	Mean TEDE (rem)	50th Percentile TEDE (rem)	95th Percentile TEDE (rem)
a	0.111	0.104	0.166
b	0.189	0.158	0.283
PAG Limits TEDE (rem)	1	N/A	5

rem - roentgen equivalent man

ENCLOSURE 3

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 2 - Total Activity Released to the Environment (Curies)

Criteria a				Criterion b			
Nuclide	**0-8 hours**	**8-24 hours**	**24-96 hours**	**Nuclide**	**0-8 hours**	**8-24 hours**	**24-96 hours**
Kr-85	{{			Kr-85	{{		
Kr-85m				Kr-85m			
Kr-87				Kr-87			
Kr-88				Kr-88			
Xe-133				Xe-133			
Xe-135				Xe-135			
Xe-135m				Xe-135m			
Cs-134				Cs-134			
Cs-136				Cs-136			
Cs-137				Cs-137			
Rb-86				Rb-86			
Rb-88				Rb-88			
Ba-139			}} Proprietary Information	Ba-139			}} Proprietary Information

ENCLOSURE 3

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 2 - Total Activity Released to the Environment (Curies)

Criteria a				Criterion b			
Nuclide	**0-8 hours**	**8-24 hours**	**24-96 hours**	**Nuclide**	**0-8 hours**	**8-24 hours**	**24-96 hours**
Ba-140	{{			Ba-140	{{		
Sr-89				Sr-89			
Sr-90				Sr-90			
Sr-91				Sr-91			
Sr-92				Sr-92			
Ba-137m				Ba-137m			
I-131				I-131			
I-132				I-132			
I-133				I-133			
I-134				I-134			
I-135				I-135			
Sb-127				Sb-127			
Sb-129			}} Proprietary Information	Sb-129			}} Proprietary Information

ENCLOSURE 3

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 2 - Total Activity Released to the Environment (Curies)

Criteria a				Criterion b			
Nuclide	**0-8 hours**	**8-24 hours**	**24-96 hours**	**Nuclide**	**0-8 hours**	**8-24 hours**	**24-96 hours**
Te-127	{{			Te-127	{{		
Te-127m				Te-127m			
Te-129				Te-129			
Te-129m				Te-129m			
Te-131m				Te-131m			
Te-132				Te-132			
Te-131				Te-131			
Rh-105				Rh-105			
Ru-103				Ru-103			
Ru-105				Ru-105			
Ru-106				Ru-106			
Rh-103m				Rh-103m			
Rh-106			}} Proprietary Information	Rh-106			}} Proprietary Information

Table 2 - Total Activity Released to the Environment (Curies)

Criteria a				Criterion b			
Nuclide	**0-8 hours**	**8-24 hours**	**24-96 hours**	**Nuclide**	**0-8 hours**	**8-24 hours**	**24-96 hours**
Nb-95	{{			Nb-95	{{		
Co-58				Co-58			
Co-60				Co-60			
Mo-99				Mo-99			
Tc-99m				Tc-99m			
Nb-97				Nb-97			
Nb-97m				Nb-97m			
Ce-141				Ce-141			
Ce-143				Ce-143			
Ce-144				Ce-144			
Np-239				Np-239			
Pu-238				Pu-238			
Pu-239			}} Proprietary Information	Pu-239			}} Proprietary Information

ENCLOSURE 3

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 2 - Total Activity Released to the Environment (Curies)

Criteria a				Criterion b			
Nuclide	**0-8 hours**	**8-24 hours**	**24-96 hours**	**Nuclide**	**0-8 hours**	**8-24 hours**	**24-96 hours**
Pu-240	{{			Pu-240	{{		
Pu-241				Pu-241			
Zr-95				Zr-95			
Zr-97				Zr-97			
Am-241				Am-241			
Cm-242				Cm-242			
Cm-244				Cm-244			
La-140				La-140			
La-141				La-141			
La-142				La-142			
Nd-147				Nd-147			
Pr-143				Pr-143			
Y-90			}} Proprietary Information	Y-90			}} Proprietary Information

ENCLOSURE 3

Supplemental Information Regarding Emergency Planning Exemption Requests in Part 6 of the ESPA

Table 2 - Total Activity Released to the Environment (Curies)

Criteria a				Criterion b			
Nuclide	**0-8 hours**	**8-24 hours**	**24-96 hours**	**Nuclide**	**0-8 hours**	**8-24 hours**	**24-96 hours**
Y-91	{{			Y-91	{{		
Y-92				Y-92			
Y-93				Y-93			
Y-91m				Y-91m			
Pr-144				Pr-144			
Pr-144m			}} Proprietary Information	Pr-144m			}} Proprietary Information

Enclosure 4 to Letter CNL-18-046

NuScale Power, LLC Affidavit, AF-0318-59303

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying RAI response, CNL-18-046, submitted by the Tennessee Valley Authority (TVA), reveals distinguishing aspects about the method by which NuScale develops its consequence analyses of postulated accidents.

NuScale has performed significant research and evaluation to develop this information and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosure 2 to TVA letter entitled "Response to Portion of Request for Additional Information Related to Emergency Planning Exemption Requests in Support of Early Site Permit Application for Clinch River Nuclear Site," CNL-18-046The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 26, 2018.

Zackary W. Rad