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

CNL-18-020

March 9, 2018

10 CFR 52.17

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

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.

The purpose of this letter is to provide the TVA response to a portion of eRAI 9206. The enclosure to this letter provides response to Key Issue 2: Questions 2 through 5 of eRAI 9206. The responses are informed by the discussions and information shared with the staff over the course of the audit.

TVA requests additional time beyond the NRC requested 30 days for the remaining RAI response, i.e., Key Issue 1, Question 1, and estimates that this response will be provided by April 2, 2018. The reason for the extension request is due to the detailed nature of the information requested for eRAI 9206, Key Issue 1. The additional time will allow TVA to complete a comprehensive evaluation of vendor source term information necessary to respond to Key Issue 1 of eRAI 9206.

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 9th day of March 2018.

Respectfully,

J. W. Shea
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Enclosure:

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 2, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

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Enclosure to Letter CNL-18-020

TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 2, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

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TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 2, 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 2: It is unclear that TVA followed the methodology in SSAR 13.3.3 with respect to the information provided about the NuScale design. In order to complete its review, the staff requires the following additional information about implementation of the SSAR 13.3.3 plume exposure pathway EPZ size technical basis methodology described in the referenced RAI response:

NRC RAI Key Issue 2, Question 2

2. Please explain how TVA followed the methodology in SSAR 13.3.3 with respect to the NuScale design information provided in the RAI response.

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 Site ESPA SSAR Subsection 13.3.3 for plume exposure pathway (PEP) emergency planning zone (EPZ) can be met. NuScale design information was used for the example analyses in a manner consistent with the SSAR Subsection 13.3.3 methodology to show the dose at the CRN Site Boundary EPZ would meet the early phase Environmental Protection Agency (EPA) Protective Action Guide (PAG) limits.

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The methodology presented in SSAR Subsection 13.3.3 discusses the development of three technical criteria from NUREG-0396, *Planning Basis for The Development of State and Local Government Radiological Emergency Response Plans In Support of Light Water Nuclear Power Plants*, for determining the required distance for the EPZ boundary. The three criteria are:

a. The EPZ should encompass those areas in which projected dose from design basis accidents (DBAs) could exceed the EPA PAGs.

b. The EPZ should encompass those areas in which consequences of less severe core melt accidents could exceed the EPA PAGs.

c. The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe core melt accidents.

From SSAR Subsection 13.3.3, the EPA PAG limit is 1 roentgen equivalent man (rem) total effective dose equivalent (TEDE) using 50th percentile meteorology. The substantial reduction in early health effects assessed by confirming the conditional probability of exceeding 200 rem (whole body acute) outside the EPZ is less than 1E-3 per reactor year. TVA followed the methodology in SSAR Subsection 13.3.3 with respect to the NuScale design information provided in the response to eRAI 8885 in Reference 1. Detailed comparison of how the methodology for the DBAs (Criterion a), less severe accidents (Criterion b) and more severe accidents (Criterion c) were followed is presented below.

<u>Criterion a - Design Basis Accident Analysis</u>

SSAR Subsection 13.3.3.1.1, *Environmental Protection Agency Protective Action Guides*, discusses four steps for verifying the dose consequences beyond EPZ do not exceed PAG limits for DBAs. The four steps are:

- Step 1 - Select appropriate accident scenarios;

- Step 2 - Determine source terms for selected accidents scenarios (source term in this context refers to fission product release to the environment as a function of time);

- Step 3 - Calculate the dose consequences for selected accident scenarios at the PEP EPZ boundary; and

- Step 4 - Compare the dose consequences for selected accident scenarios with the EPA PAG.

For Step 1, the DBAs are identified in the postulated accidents defined in the FSAR Chapter 15 of the combined license application (COLA). The NuScale Design Certification Application (DCA) Chapter 15 information was used for the example analysis, in lieu of COLA information, since TVA has not yet selected a reactor design. A large break loss of coolant accident (LOCA) is considered the reasonably bounding DBA. However, for the NuScale design, the radionuclide release from a design basis LOCA is from coolant activity, with no release from the fuel. As a result, for the NuScale example analysis summarized in Reference 1, a surrogate source term was used which is the design basis source term (DBST) based on beyond design basis accidents.

Consistent with Step 2 of the SSAR Subsection13.3.3 methodology, the source term for DBAs used in the example analysis was calculated based on the NuScale Chapter 15 information. For Step 3, calculation of TEDE in the NuScale example analyses was performed using the specific dose pathways described in SSAR Subsection 13.3.3.1.1, i.e., cloudshine, inhalation, ground and resuspension. Consistent with Step 4, the resulting dose consequence for the DBST in the NuScale example analyses was confirmed not to exceed the EPA PAG limit at the CRN PEP EPZ, i.e., CRN Site boundary. Therefore, in the NuScale example, the Criterion a design basis accident analysis was conducted consistent with the methodology described in SSAR Subsection 13.3.3.

Criterion b - Less Severe Accident Analysis

The four steps described in SSAR Subsection 13.3.3.1.1 for Criterion a DBAs also apply for Criterion b less severe accident analyses. Details on how Criterion b less severe accidents were analyzed in the NuScale example analyses are described below.

For Step 1 of the for Criterion b analysis, the accident selection is based on two mean core damage frequency (CDF) limits. The first CDF limit of greater than 1E-8 per reactor year is used to initially select accident sequences for consideration. The selected sequences are then grouped in accident scenarios and only scenarios with a mean CDF greater than 1E-6 per reactor year are selected. An intact containment is a specified requirement for an accident to be classified as a Criterion b accident. The selection of the less severe accident scenarios developed to conduct the example analyses was informed by NuScale Probabilistic Risk Assessment (PRA) results. The NuScale PRA has a total mean CDF of 9.2E-8 per reactor year (Reference 3, Table 19.1-80). Therefore, regardless as to whether individual sequences or grouped scenarios are considered, there are no accident scenarios with a mean CDF greater than 1E-6 per reactor year. Based on this CDF threshold for the NuScale design, there are no less severe accidents that screen-in for Criterion b analysis. However, there is an additional requirement in the SSAR Subsection 13.3.3.1.1 methodology (footnote), if no scenarios are identified for a CDF greater than 1E-6, then a suitable surrogate source term and release sequence will be developed and justified. To comply with this additional requirement, the NuScale example analysis uses one of the sequences which comprise the DBST as the suitable surrogate to be applied to Criterion b. This accident is a loss of direct current (DC) power with incomplete emergency core cooling system (ECCS) actuation. Justification for selecting this DBST accident sequence as the suitable surrogate for Criterion b is provided in the response to RAI question 3 below.

For Steps 2 and 3, the environmental release based on the release fraction into containment and evaluation of the resulting dose consequences was conducted in the same manner as explained above for Criterion a. Consistent with Step 4 of the SSAR Subsection 13.3.3 methodology, the dose consequence for the selected DBST was confirmed not to exceed the EPA PAG limit at the selected PEP EPZ i.e., CRN Site Boundary. Therefore, in the NuScale example, the Criterion b less severe accident analysis was conducted consistent with the methodology described in SSAR Subsection 13.3.3.

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Criterion c - More Severe Accident Analysis

SSAR Subsection 13.3.3.1.2, *Substantial Reduction in Early Health Effects*, describes five steps for verifying that areas outside of the PEP EPZ meet the limits for substantial reduction in early health effects based on more severe accidents. The five steps are:

- Step 1 - Select appropriate accident scenarios.
- Step 2 - Determine source terms for selected accident scenarios (source term in this context refers to fission product release to the environment as a function of time).
- Step 3 - Calculate the dose consequences for selected accident scenarios at the PEP EPZ boundary.
- Step 4 - Calculate the distance at which the conditional probability to exceed 200 rem (whole body) exceeds 1E-3 per reactor year.
- Step 5 - Compare that distance with the PEP EPZ.

For Step 1, the accident selection criteria for Criterion c more severe accidents are the same as Criterion b, except that the CDF limit for Criterion c grouped scenarios is greater than 1E-7 per reactor year, compared to the greater than 1E-6 per reactor year limit for Criterion b. The grouped accident scenarios with a mean CDF greater than 1E-7 per reactor year encompass the more severe core damage accident scenarios. In the example analysis conducted using the NuScale design; no accidents were identified that remain screened-in past Step 1 of the Criterion c methodology as described in SSAR Subsection 13.3.3.1.2. During Step 1 implementation, one accident with a CDF greater than 1E-8 was identified, but this accident is a unique accident with no similar sequences. Therefore, this accident could not be grouped with other similar sequences into scenarios. The sequence stands alone as a scenario, is less frequent than 1E-7 per reactor year and is therefore screened-out. Further explanation of this accident is provided in the response to RAI question 5 below. Therefore, in the NuScale example, the Criterion c more severe accident analysis was conducted consistent with the methodology described in SSAR Subsection 13.3.3.

At the COLA stage, based on the selected reactor design, if any accidents are selected as a result of Step 1, the methodology to verify that areas outside of the PEP EPZ meet the limits for substantial reduction in early health effects based on more severe accidents as described in SSAR Subsection 13.3.3.1.2 would be followed.

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TVA Response to NRC Electronic Request for Additional Information (eRAI) 9206, Key Issue 2, Related to Emergency Planning Exemption Requests in Part 6 of the ESPA

NRC RAI Key Issue 2, Question 3

3. With respect to the more frequent less severe accidents, provide the analysis describing consideration of severe accidents other than those used to develop the design basis source term or justify why it is not necessary to perform such an analysis.

TVA Response

The accidents that comprise the DBST in the example NuScale analysis are representative of the full scope of the less severe accidents from the full power, internal events PRA and are inclusive of all hazard models. In the example analysis summarized in Reference 1, to comply with the SSAR Subsection 13.3.3 methodology, a surrogate source term and release sequence were developed, since there were no accident scenarios in the NuScale design with a CDF greater than 1E-6 per reactor year. The NuScale example analysis uses one of the DBST accident sequences as the suitable surrogate to be applied to Criterion b. This accident is a loss of DC power with incomplete ECCS actuation. Selection of the DBST loss of DC power accident for the Criterion b less severe accidents in the example analyses was based on event frequency. In the considered accident, a loss of DC power leads to an immediate actuation signal for ECCS valves. There is a temporary block of ECCS actuation due to high reactor pressure vessel pressure, but once pressure drops due to successful decay heat removal system operation, the reactor vent valves open upon demand while the reactor recirculation valves fail to open. This event has the highest frequency of the sequences used to create the DBST, due to the initiating event and ECCS failure mode. This event progression also represents the top individual cutset (sets of failures that result in overall failure) for full power internal events (Reference 3, Table 19.1-18). Additionally, the most frequent individual loss of DC power sequence contributes approximately 16 % of the total full power internal events CDF (Reference 3, Table 19.1-17). From a frequency perspective, the analysis of accidents that informed the DBST and the selection of the loss of DC power is justified as the appropriate surrogate for less severe accident analysis.

NRC RAI Key Issue 2, Question 4

4. With respect to the severe accident scenario selection in general, contrary to the methodology implementation discussion in SSAR 13.3.3.1.4, the analysis did not discuss all relevant plant states (i.e., the scenario selection only included full power events, and did not include discussion of low power and shutdown events) and did not consider external hazards. Provide this analysis or justify deviating from the SSAR 13.3.3.1.4 information on implementation of the SSAR plume exposure pathway size basis methodology.

TVA Response

The example NuScale analyses conducted to support the eRAI 8885 response did not deviate from the SSAR Subsection 13.3.3.1.4 methodology for implementation of PEP EPZ sizing. As described in SSAR Subsection 13.3.3.1.4, once the reactor design is selected at the COLA stage, the COLA PRA will address the applicable plant events including full power, low power and shutdown, design-specific internal and site-specific external events.

The NuScale PRA results in the DCA also include CDF values for all internal and external events and all power states and operating modes. The selection of the severe accident scenarios developed to conduct the example analyses was informed by NuScale full power internal events PRA results. In the NuScale PRA external hazards models, external flooding and high wind events result in a loss of AC offsite power (Reference 3, Tables: 19.1-57, 19.1-62, and 19.1-63). Both external flooding and high winds have a larger total CDF than full power, internal events (Reference 3, Table 19.1-80). However, a loss of onsite DC power accident has a shorter time to core damage, compared to a loss of offsite power. This is because a loss of offsite AC power allows at least 24 hours of battery capacity to delay ECCS actuation, while a loss of DC power actuates the ECCS immediately. Compared to the full power CDF, the CDF for low power shutdown is very low except for the crane failure accident (Reference 3, Table 19.1-80). The crane failure accident was considered in the Criterion c analysis. Further explanation of this accident is provided in the response to RAI question 5 below. When considering low power, shutdown events and external hazards, the loss of DC power accident scenario analyzed in the example analysis is justified as the appropriate surrogate Criterion b analysis.

NRC RAI Key Issue 2, Question 5

> *5. The staff notes that the analysis does not appear to consider the beyond design basis event with highest risk described in the NuScale design certification application Environmental Report (ADAMS Accession No ML17013A296). The reactor building crane failure accounts for 99% of total CDF, its CDF is two orders of magnitude larger than the next highest release category and the source term fraction of core released is larger. Considering the discussion in SSAR 13.3.3, including the implementation information in 13.3.3.1.4, discuss whether this event would be included in severe accident scenario selection and Please explain why this event was not considered in the analysis provided to support the RAI response.*

TVA Response

SSAR Subsection 13.3.3.1.2, *Substantial Reduction in early Health Effects*, describes the methodology for verifying that areas outside of the CRN Site PEP EPZ meet the limits for substantial reduction in early health effects. The accident scenario selection, i.e., Step 1 of the methodology, is described as follows:

- Step 1 - Accident Scenario Selection: More severe core melt accident scenarios include postulated containment failure/bypass accidents with the potential for higher consequences with mean CDF greater than 1E-7 per reactor year. In performing the accident sequence grouping as part of scenario selection, accident sequences with mean CDF greater than 1E-8 per reactor year should be considered in the initial sequence selection.

The NuScale crane failure accident, which conservatively is assumed to result in a failure of containment, was considered in the example analysis conducted to support eRAI 8885 response. For the NuScale design, the CDF for a crane failure accident (also known as a dropped module accident) is 8.8E-8 (Reference 3, Table 19.1-80). Therefore, if taken as a single sequence the crane failure accident would be considered in the initial sequence selection. However, a crane failure accident for the NuScale design is a unique event where there are no

similar sequences that may be used for grouping. Therefore, as an accident scenario, a crane failure accident is not greater than the scenario frequency limit of 1E-7 per reactor year and is screened-out. As described in SSAR Subsection 13.3.3.1.4, *COLA*, applicable plant operating states (modes) including full power, low power and shutdown, and design-specific operating states unique to the selected design (e.g., refueling and/or concurrent power operation and refueling) will be evaluated at COLA.

At the COLA stage, based on the selected reactor design, TVA will apply the methodology described in SSAR Subsection 13.3.3.1.1 for EPA PAG and SSAR Subsection 13.3.3.1.2 for substantial reduction in early health effects to the selected reactor design. At COLA, if a design-specific accident is identified and found to be credible (i.e., screened-in) consistent with the methodology described in SSAR Subsection 13.3.3 it would be included in accident scenario selection and evaluated for dose consequence.

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. NuScale Power, LLC, Design Certification Application, "Chapter 15: Transient and Accident Analyses," Revision 0, December 2016

3. NuScale Power, LLC, Design Certification Application, "Chapter 19: Probabilistic Risk Assessment," Revision 0, December 2016