RPAC Technical Issues for Discussion
Public Meeting with NuScale Power, LLC
April 18, 2017

Failed Fuel Fraction

10 CFR 52.47(a)(5) requires applicants to identify the kinds and quantities of radioactive materials expected to be produced during operations and the means for controlling and limiting radiation exposures. 10 CFR Part 20 requires the use of engineering features to control and minimize the amount of radiation exposure to members of the public and occupational workers, from both internal and external sources. 10 CFR 50.49(e)(4) requires applicants to identify the type of radiation and the total dose expected during normal operation over the installed life of the equipment. General Design Criterion (GDC) 4 of Appendix A to 10 CFR Part 50 requires applicants to ensure that structures, systems, and components (SSC) important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation.

Design Specific Review Standard (DSRS) section 12.2 Acceptance Criteria states in part, that the shielding and ventilation design fission product source terms will be acceptable if developed using the bases of 0.25-percent (%) fuel cladding defects (aka design basis failed fuel fraction value) for pressurized-water reactors (PWRs) or the reactor coolant system (RCS) isotopic concentrations, including fission products and significant corrosion and activation products, equivalent to operation for a full fuel cycle at the technical specification (TS) limits for halogens (I-131 dose equivalent) and noble gases (Xe-133 dose equivalent). DSRS Chapter 11, Sections 11.1, 11.2, and 11.3; NUREG-0800 “Standard Review Plan” (SRP) Chapter 3, Section 3.11; and Branch Technical Positions (BTPs) 11-5 and 11-6 also provide guidance on fuel leakage (failed fuel fraction) assumptions.

NuScale Design Control Document (DCD), Tier 2, Revision 0, Chapter 11, Table 11.1-2: “Parameters Used to Calculate Coolant Source Terms,” shows that the design basis failed fuel fraction value is 0.028% (which is proposed as the basis for determining plant radiation shielding, zoning, ventilation design, equipment qualification (EQ) dose calculations, etc).

NuScale Technical Report TR-1116-52065 Revision 0 “Effluent Release (GALE Replacement) Methodology and Results,” describes the derivation of the 0.028% value. In essence, the method determined the **average** fuel failure fraction over a multiple year period (0.0028%), and then multiplied that value by a factor of 10, to arrive at the proposed 0.028% failed fuel fraction value to be used for shielding and ventilation system design, etc. The value of 0.028%, is less than one tenth of the TSs section 3.4.8 RCS Specific Activity Limit of 0.2 micro Ci/gram Dose Equivalent Iodine (DEI).

NuScale used data from “Benchmarking of GALE-09 Release Predictions Using Site Specific Data from 2005 to 2010,” PNNL-22076, dated November 2012, to determine a realistic fission product source term to be used for the evaluation of normal effluent releases. This report examined the reported average fuel failure fraction from 2005 to 2010. DCD subsection 11.1 states “(t)he design basis source term assumes a conservative value of equivalent fuel defects an order-of-magnitude greater than the realistic coolant source term. This results in a design basis failed fuel fraction that is ten times greater than the realistic failed fuel fraction.” In the view

of the staff, an empirical survey of operational experience regarding failed fuel experiences at operating reactor facilities does not constitute a safety case for the proposed NuScale failed fuel fraction of 0.028%. Further, the use of an average failed fuel experience, without a corresponding Technical Specification limit for DEI and Dose Equivalent Xenon, does not comport with established licensing practices used by the staff for evaluating the acceptability of the proposed design bases for shielding and ventilation systems. The value proposed by NuScale is not a bounding design bases value, since it is about a factor of 5 less than the amount of fuel cladding defects that occurred at one plant in 2009, which had been designed for normal system operation with clad defects in fuel rods generating 1% of rated core thermal power. The NuScale proposed value is comparatively less than that used by other plant designs, including passive designs, and is less than the historically used 0.25% failed fuel fraction, is not bounding with respect to operational data within the stated time frame, and is not conservative. Based on the information provided, this lower failed fuel fraction would not meet the acceptance criteria in NuScale DSRS Chapter 11, Sections 11.1, 11.2, 11.3, and Chapter 12, Section 12.2; NUREG-0800 Chapter 3, Section 3.11; and BTPs 11-5 and 11-6. (Note: for shielding and ventilation system design, the value is about a factor of 10 less than that used by the NRC to license plants under 10 CFR Parts 50 and 52, and about a factor of 10 less than the value currently specified in NuScale DCD TS Part 4, Volume 1, Section 3.4.8.)

Based upon the docketed information, the NRC staff is unable to determine whether the NuScale design is adequate to protect members of the public and occupational workers from exposure to radiation and protection of SSCs important to safety. The staff requests the applicant to provide additional information (e,g., requisite analyses and safety margin evaluations) that clearly demonstrates, through the implementation of the shielding and ventilation system acceptance criteria stated in DSRS 12.2 and DSRS 11.1, that the NuScale design provides reasonable assurance that the public and occupational workers will be protected from exposure to radiation. Explain why the proposed assumed failed fuel fraction is appropriately conservative for the purposes of evaluating personnel doses, radiation protection design features, radwaste handling system capacities, and equipment qualification analyses. Explain why adopting a technical specification limit that bounds the newly proposed failed fuel fraction is not warranted as discussed in the DSRS acceptance criteria stated above.

Radiation Protection Shielding Design

10 CFR 52.47(a)(8) requires that the final safety analysis report provide the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

10 CFR 50.34(f)(2)(vii) requires that applicants perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

10 CFR 50.34(f)(2)(viii) requires that applicants provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.

NUREG-0737 and DSRS section 12.3-12.4 provide additional guidance on acceptable methods of meeting these requirements. These documents indicates that post accident radiation zones should be provided based on the guidance of RGs 1.7 and 1.183 and that the analysis for access to vital areas should consider access to, stay time in, and egress from these vital areas. NUREG-0737 specifies that any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is to be designated as a vital area and in addition to the control room and technical support center, the sample station and sample analysis area must be included among those areas where access is considered vital after an accident (This question is focused on the sample station, sample analysis, and other areas requiring infrequent access. Any questions related to the MCR and TSC will be addressed separately). Finally, NUREG-0737 provides a list of other areas that should be considered in determining the vital areas. NUREG-0737 specifies that if these areas are not considered vital areas, justification should be provided for not including them. The areas specified are the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if applicable), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. In addition, any other areas that may need to be accessed during an accident are to be identified. As specified, the plant should be designed so that the dose to an individual should not exceed the occupational dose criteria to perform the vital missions, including accessing and egressing from the areas.

DCD Section 12.4.1.8, “Post-Accident Actions,” provides a discussion of post-accident sampling and analysis for both primary liquid sampling and containment gas sampling. It indicates that access may be required to the CVCS gallery, counting room, and hot lab in the Reactor Building, and Annex building counting room may be required to sample and analyze liquid samples and that access to the utilities area and steam gallery on the 100’ elevation of the reactor building may be required for containment gaseous grab sampling. The discussion to perform these activities indicates that doses will be under the 5 rem occupational dose limit, including ingress and egress. However, it indicates that 0.25” lead equivalent temporary shielding is assumed in the analysis for calculating the post-accident doses (in addition to the permanent shielding specified in the application) and there is no discussion of the dose rates received in installing the temporary shielding to the various different areas. In addition, no post-accident radiation zone maps or ingress/egress routes are provided; no information is provided on the time assumed to be spent at each location or the speed of travel assumed; and no information is provided on the assumptions made regarding submersion and inhalation dose. Finally, staff did not find any information regarding any other vital areas in the DCA.

Based on the above, staff requests the following:

1. Provide more information regarding where temporary lead equivalent shielding is assumed to be installed as it relates to taking and analyzing liquid and gaseous samples, including access and egress, and the dose rates that will be received in installing this temporary shielding during accident conditions.

2. Provide post-accident radiation zone maps and ingress and egress routes for all areas associated with performing the vital missions and update the DCD to include this information.

3. Provide information on the time spent at each location and the speed of travel assumed and update the DCD to include this information.

4. Provide information on the assumed post-accident airborne activity concentrations and how these source terms were developed. In addition, provide information regarding what assumptions are made regarding submersion and inhalation dose. Update the DCD accordingly.

5. Identify any other areas that would require or may require access following an accident to permit an operator to aid in the mitigation of or recovery from an accident. For each of these areas, identify the work that may need to be performed in these areas, provide the dose to perform the work and to access and egress from the area. Also, for each of these areas, provide the information requested in items 1 through 4, as applicable. Update the DCD with this information, as appropriate.

6. If any of the areas that NUREG-0737 specifies should be included for consideration as a vital area (i.e. hydrogen control system, containment isolation reset control area, manual ECCS alignment area, motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels), are not considered as a vital area in the NuScale design, please provide justification for not including them. The justification should include an explanation for why it will not be necessary to access each of the areas following an accident.

<u>Maximum Hypothetical Accident</u>

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 10 CFR 52.47(a)(2)(iv)(A) and 10 CFR 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents (DBAs) are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale DSRS 15.0.3.

Note that the referenced topical report cited in DCD Section 15.0.3.3. does not give the values for the (MHA) DBST source term release fractions, as implied in the last sentence of the text of this DCD Section. It only gives the method and example calculation.

Consequently, the staff requires the following information to conduct its review of the maximum hypothetical accident (MHA):

1. What accident scenario(s) are used to develop the design basis source term (DBST) for the MHA?

a. Referenced topical report, TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," gives the methodology to choose scenarios – what was result of the use of the methodology?

b. Are the severe accident scenarios used to develop the DBST the same ones evaluated in the PRA?

c. Describe the accident scenario(s) that are used to develop the DBST, including initiating event and progression.

2. On DCD page 15.0-33 it states that core release fractions to containment are given in the referenced topical report. Where exactly in the topical report? This statement appears to be in conflict with various scope statements made in the topical report such as the following:

- The topical report gives example release fractions related to example scenarios. In the second paragraph of Section 1.2 "Scope" of the topical report, it states that "This topical report is not intended to provide final DBST isotopic inventory values, final dose values, final atmospheric dispersion factors, or final values of any other associated accident evaluation; rather, example values for the various evaluations are provided for illustrative purposes."

- Topical report Section 4.2.3, "Release Timing and Magnitude," describes the method for how the release timing and magnitude (fraction of core inventory) will be determined from the MELCOR calculations of candidate "source term DBAs," but does not provide values for the core release fractions and their timing.

- Section 5.0, "Example Calculation Results," of the topical report states that the example calculation analyses and results are presented to demonstrate the application of the methodology, and that "NuScale plans to provide the final design values in the design certification application."

3. What are the aerosol removal rates assumed in the DBA dose analyses?

4. DCD Tier 2 Tables 12.2-28, 12.2-29, and 12.2-30 look like they give MHA analysis parameters in discussing the post-accident source used in shielding analyses. If so, these tables should at least be referenced from DCD Tier 2 Section 15.0.3.9 or it may be more appropriate to move the information to Chapter 15.

Control Habitability

Criteria on control room habitability, including dose to operators during accidents, are provided in 10 CFR Part 50, Appendix A, GDC 19. The design basis accident (DBA) dose analyses in DCD Tier 2 Chapter 15 were performed, in part, to show compliance with GDC 19.

In DCD Tier 2 Section 15.0.3.7.1 it states that the control room ventilation system design modeling assumptions are given in the referenced topical report, TR-0915-17565-P, Rev.1, "Accident Source Term Methodology." Although not specifically referenced, Section 3.3.4.1 of the topical report provides some information on control room ventilation design. Table 3-4 of the topical report is titled "Example control room characteristics," and, as noted in the text in Section 3.3.4.1, is only intended to provide the values used in the example calculations provided in the topical report. Information on the control room characteristics relevant to radiological protection and used in the estimation of dose to control room operators is also not evident in DCD Tier 2, Sections 6.4 and 9.4.1. Furthermore, the example values in the referenced topical report give much different values for component flow rates than those given in DCD 6.4 and 9.4.1.

In order to complete its review of the applicant's evaluation of the DBA control room radiological habitability, the staff requires additional information. Provide the following information used as assumptions and inputs to the applicant dose analyses that support the evaluations in Sections 6.4 and 15.0.3 of the DCD and make revision to the DCD to document:

- Control room envelope (CRE) volume
- Normal control room HVAC (heating, ventilation and air conditioning) system (CRVS) normal ventilation intake, unfiltered flow rate
- CRVS post-accident supplemental filtration mode intake flow rate (if supplemental filtration credited after 72 hours)
- CRVS recirculation flow rate and whether recirculation flow is filtered
- Intake radiation monitor setpoints for control room isolation and initiation of the control room habitability system (CRHS) and for initiation of the CRVS supplemental filtration mode
- CRHS initiation time based on intake radiation high signal, per DBA