NuScaleDCRaisPEm Resource

From: Cranston, Gregory
Sent: Thursday, November 30, 2017 8:33 AM
To: RAI@nuscalepower.com
Cc: NuScaleDCRaisPEm Resource; Lee, Samuel; Chowdhury, Prosanta; Karas, Rebecca; Burja, Alexandra; Markley, Anthony; Schmidt, Jeffrey
Subject: RE: Request for Additional Information No. 290 RAI No. 9157 (9.1.1)
Attachments: Request for Additional Information No. 290 (eRAI No. 9157).pdf

Attached please find NRC staff's request for additional information concerning review of the NuScale Design Certification Application.

Please submit your technically correct and complete response within 60 days of the date of this RAI to the NRC Document Control Desk. The NRC Staff recognizes that NuScale has preliminarily identified that the response to this question in this RAI is likely to require greater than 60 days.

NuScale is expected to provide a schedule for the RAI responses by email within 14 days.

If you have any questions, please contact me.

Thank you.

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Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
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301-415-0546

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Request for Additional Information No. 290 (eRAI No. 9157)

Issue Date: 11/30/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling
Application Section: 9.1.1

QUESTIONS

09.01.01-12

Standard Review Plan (SRP) Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 62, “Prevention of criticality in fuel storage and handling,” and 10 CFR 50.68, “Criticality accident requirements.” SRP Section 9.1.1 instructs the reviewer to verify that the fuel storage rack data are complete and that the criticality analysis conservatively incorporates fuel storage rack design data. Furthermore, SRP Section 9.1.1 guides the reviewer to evaluate the normal- and abnormal-conditions models to verify that normal and abnormal conditions are modeled correctly and that all modeling approximations and assumptions are appropriate.

During its audit of documents related to FSAR Section 9.1.1 and of criticality analysis input files, the staff noted the following apparent errors that have potential small nonconservative effects on the analysis results:

a. The input files for damaged fuel (files sc-ns-dm-1 through -10, generated in response to RAI 8760, Question 09.01.01-1) appear to use the minimum storage tube thickness allowed by tolerances, not the nominal value as they should. Per the tolerance analysis in TR-0816-49833, “Fuel Storage Rack Analysis,” the minimum storage tube thickness has a small nonconservative effect.
b. The above input files for damaged fuel, except for the all-damaged-fuel cases, have one normal (undamaged) fuel rod in each damaged fuel assembly, while TR-0816-49833 states that a damaged fuel assembly is assumed to have a cladding failure in 100 percent of the fuel rods.
c.. The input files for damaged fuel cases have only one latticecell card that specifies the gaps as voids for cross-section processing. In reality, the gaps are flooded with water for damaged fuel.

In addition, the staff noted that Section 5.8 of EC-B160-3283, “Spent Fuel Storage Rack Criticality Analysis,” Revision 0, describes how the criticality analyses incorrectly used a rack-to-rack spacing that was too large. The document includes justification that the differences in k-effective are small and do not affect the results or conclusions of the calculation. The staff notes that the spacing error, in combination with the potential errors above, may become more significant than assessed in EC-B160-3283.

Therefore, please address the above by (1) correcting the input files and performing new calculations, (2) evaluating the impacts of the errors and imposing an appropriate penalty, or (3) providing justification that the analysis results are still conservative. Consider the effect of combining the errors mentioned in (a) through (c) as well as the error documented in EC-B160-3283. For any approach, update TR-0816-49833 as necessary.

09.01.01-13

SRP Section 9.1.1 provides guidance for complying with GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 directs the reviewer to verify that the computational method validation description is complete and that the validation study used acceptable methods and was performed correctly.

As described in Section 3.3.7.2.6 of TR-0816-49833-P, the criticality code validation study includes a trending analysis against boron separator plate areal density. However, the staff notes that the analysis appears to have used atom density (units of atoms/b-cm), not areal density (units of atoms/b). Trending against areal density has considerably more value than trending against atom density due to the cross-sectional dependence of neutron absorbers. The staff also notes that the experiments do not use a constant plate thickness, so the current trend results for atom density would not be exactly translatable to areal density. Therefore, please re-perform the trending analysis using the correct values of areal density, and update TR-0816-49833-P accordingly. Also update the numerical NuScale value of areal density in Section 3.3.7.2.6 and Table 3-89 of TR-0816-49833-P, as the current value of 2.8E-3 appears to correspond to the atom density, not the areal density.

09.01.01-14

SRP Section 9.1.1 provides guidance for complying with GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 directs the reviewer to verify that the computational method validation description is complete and that the validation study used acceptable methods and was performed correctly. Furthermore, the applicant cites the guidance in NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology.”

Technical report TR-0816-49833-P states that there is no basis to justify a modification to the calculated bias as a result of any trend due to small slopes, poor correlation coefficients, and fitting parameter uncertainty in the linear regression. Although the uncertainties in some of the data measurements are relatively large, the staff does not consider that a sufficient basis to disregard a trend. The staff observes that there may be valid trends in enrichment, assembly separation, soluble boron content, and boron plate areal density.

Furthermore, the staff could not confirm the normality of the assembly separation and soluble boron content data sets. The staff notes that NUREG/CR-6698 specifies that non-parametric treatment should be used if normality cannot be established.

Finally, Section 3.3.7.5, “Test for Normality,” of TR-0816-49833-P indicates that the data set used for the code bias and bias uncertainty calculation is normally distributed. However, this is based on critical values for a sample size of 68 since no value is tabulated for 69. The staff observes that the critical values can be linearly interpolated, and in that case, the critical values for a sample size of 69 are approximately 156.6 < D’ < 164.5. Using interpolation, the D’ value is no longer between the critical values, which means there is not a 95 percent probability that the data set is normally distributed.

Therefore, please do the following:

a. Implement a conservative trending analysis, justify how the current trending analysis is conservative, or implement a penalty to cover for the lack of consideration of trends in the code bias.
b. If a conservative trending analysis is to be implemented, use the more conservative of the trended and untrended bias and bias uncertainty in calculating the final code bias and bias uncertainty.
c. If the untrended bias and bias uncertainty is to be used, provide further justification that the data set is normally distributed, or alternatively, use a non-parametric statistical treatment.
d. Provide the test statistics and the critical values for the normality tests in TR-0816-49833-P.
e. Update TR-0816-49833-P accordingly.

09.01.01-15

SRP Section 9.1.1 provides guidance for complying with GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 directs the reviewer to verify that the computational method validation description is complete and that the validation study used acceptable methods and was performed correctly. Furthermore, the applicant cites the guidance in NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology."

The staff notes that the applicant used water and concrete albedos in its criticality analysis models. However, it is not clear whether those albedos were used in the code validation. Per NUREG/CR-6698, "calculations made for actual criticality safety analyses should not use code options (e.g., albedo, biasing, boundary conditions, etc.) that are dissimilar from those used in the validation. These code options incorporate approximations of the code response. Unless these options are also validated their use is not appropriate."

Therefore, please describe how the concrete and water albedos were validated. If they were not validated, either revise the models in the code validation to include the albedos or justify the lack of validation. Finally, update TR-0816-49833-P accordingly.

09.01.01-16

If credit for soluble boron is taken, 10 CFR 50.68 requires that k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level (95/95), if flooded with borated water, and the k_{eff} must remain below 1.0 (subcritical), at a 95/95, if flooded with unborated water. SRP Section 9.1.1 provides guidance for complying with 10 CFR 50.68 and states that k_{eff} must include allowance for all relevant uncertainties and tolerances.

Equation 1 of TR-0816-49833-P is the governing equation to calculate k_{eff} at 95/95. It is not clear to the staff why the confidence multiplier *C* is not applied to the standard deviation of the sum of statistically independent manufacturing tolerances while it is applied to all other standard deviations in the equation. Therefore, either justify why *C* is not applied to the standard deviation associated with manufacturing tolerances, or update TR-0816-49833-P to apply it thusly.

09.01.01-17

SRP Section 9.1.1 provides guidance for complying with GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 directs the reviewer to evaluate the identification of abnormal conditions to verify that the scope of considered abnormal conditions is comprehensive and to evaluate the abnormal-conditions models to verify that abnormal conditions are modeled correctly and that all modeling approximations and assumptions are appropriate.

FSAR Tier 2, Section 9.1.1 discusses a boron dilution event qualitatively, stating that the large volume of water in the spent fuel pool prevents undetected addition of unborated water sufficient to dilute the boron concentration from the minimum required by technical specifications to the 800 ppm credited in the criticality analyses. However, no quantitative analyses are performed. Please provide additional justification for the statements in the FSAR regarding boron dilution, such as the time required to dilute the pool to 800 ppm versus the amount of time required for an action to be taken to prevent further dilution, and update the FSAR accordingly.

09.01.01-18

GDC 62 requires criticality in the fuel storage and handling system to be prevented by physical systems or processes, preferably by use of geometrically safe configurations. 10 CFR 50.68 defines the limits on k-effective for new and spent fuel storage. The information in the design certification application that supports meeting these regulations needs to be accurate and consistent so the staff is able to make a reasonable assurance finding.

The staff noted that Section 3.3 of TR-0816-49833-P contains several apparent typographical errors that affect technical meaning or details. These errors are listed below:

a. Section 3.3.1, "Methodology," states that the criticality analysis used the functional module NITAWL-II for cross-section processing. However, it appears from the calculation files that the staff audited that the module CENTRM/PMC was used rather than NITAWL-II.
b. Contrary to the tolerances listed for the poison plate gap in Tables 3-71 and 3-72, the staff's audit of the calculation output files for those cases showed that the minimum and maximum gaps considered were consistent with the allowances discussed in the last paragraph of Section 2.2 of audit document EC-B160-3283. This corresponds to larger increases and decreases relative to the nominal case than listed in Tables 3-71 and 3-72 of TR-0816-49833-P.
c. The staff's audit of the calculation output files for the damaged fuel analysis for corner locations described in the response to RAI 8760, Question 09.01.01-1 (files sc-ns-dm-3 and sc-ns-dm-8) considered only four damaged fuel assemblies, not five as listed in Table 3-73. Since the applicant has already analyzed a bounding case (all damaged fuel assemblies), it is not necessary to rerun the analysis using five damaged assemblies; however, Table 3-73 should be updated to reflect that only four damaged fuel assemblies were considered for the corner location analysis.
d. The markup of Section 3.3.6.2 of TR-0816-49833-P provided in the response to RAI 8760, Question 09.01.01-1 states that three scenarios with damaged fuel assemblies are simulated; however, a scenario was added, so four scenarios were simulated.
e. Section 3.3.7.2.5 of TR-0816-49833-P states that the modeled low-boron concentration for the fuel storage racks is 200 ppm; however, the actual modeled concentration is 800 ppm.

Please address the above items by either (1) updating TR-0816-49833-P to correct them or (2) justifying why the information is accurate.