NuScale Power

RAIO-0617-54653

June 26, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 04 (eRAI No. 8760) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 04 (eRAI No. 8760)," dated April 25, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 8760:

- 09.01.01-1
- 09.01.01-2
- 09.01.01-3
- 09.01.01-4
- 09.01.01-5
- 09.01.01-6
- 09.01.01-7
- 09.01.01-8
- 09.01.01-9
- 09.01.01-10
- 09.01.01-11

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 04 (eRAI No. 8760). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, TWFN-6E55
Omid Tabatabai, NRC, TWFN-6E55
Samuel Lee, NRC, TWFN-6C20

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8760, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8760, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0617-54655

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8760, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8760, nonproprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-1

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.68(b)(4) requires, for criticality analyses that take credit for soluble boron, that the k-effective 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, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

To meet 10 CFR 50.68(b)(4), the staff considers it acceptable to demonstrate through criticality analyses that:

- For normal conditions, k-effective does not exceed 0.95 when flooded with borated water and remains below 1.0 when flooded with unborated water; and
- For abnormal conditions, k-effective does not exceed 0.95 when flooded with borated water.

Technical report TR-0816-49833-P, "Fuel Storage Rack Analysis," contains a criticality analysis for damaged fuel to demonstrate that k-effective remains below 0.95 when flooded with borated water. However, the staff considers the need for the racks to hold five damaged fuel assemblies to be a design feature of the racks and therefore a normal condition, contrary to the statement in FSAR Tier 2, Section 9.1.1.3.2 that damaged fuel constitutes an abnormal condition. Therefore, provide justification that k-effective remains below 1.0 when the racks are flooded with unborated water. To demonstrate compliance with 10 CFR 50.68(b)(4), please update TR-0816-49833-P with such justification and clarify in FSAR Tier 2, Section 9.1.1.3.2 that damaged fuel is not an abnormal condition.

NuScale Response:

FSAR Section 9.1.1.3.2 has been updated as shown in the attached markups to no longer state that storage of up to 5 damaged fuel assemblies constitutes an abnormal condition.

Technical Report TR-0816-49833-P, Rev 0, Fuel Storage Rack Analysis, has been updated with the criticality analysis results showing that k-effective for damaged fuel remains below 1.0 when flooded with unborated water. Comparing the results between the cases with no damaged fuel and 5 damaged fuel assemblies in a borated moderator at 800 ppm boron, the difference in $k_{95/95}$ is small, 0.00026, as shown in the updated Table 3-73.

An additional calculation was performed that considered five damaged fuel assemblies with unborated water. The updated Table 3-73 shows that with an unborated moderator, the difference in $k_{95/95}$ results increased slightly to 0.00035, but remained within the same order of magnitude. Because the response to RAI Question 09.01.01-4 addresses criticality results at a full moderator density at 40 degrees F, the previous calculations for damaged fuel were also revised to account for changes in the bias and uncertainty associated with a full density moderator. In reviewing the previous analyses, it was identified that the racks were not completely full of fuel assemblies. As a result, the revised analyses consider full racks of fuel and a pool of unborated water at full moderator density.

Section 3.3.6.2 of TR-0816-49833 has been revised to identify that cases are analyzed for both unborated and borated water. Table 3-73 has been revised to add the cases for unborated water. The results demonstrate that the k-effective values remain less than the limiting values for both unborated and borated water.

Impact on DCA:

FSAR Section 9.1.1.3 and related Technical Report TR-0816-49833, Fuel Storage Rack Analysis, have been revised as described in the response above and as shown in the markups provided with this response.

Additional Information:

Impact on Technical Report TR-0816-49833

Section 3.3.6.2 and Table 3-73 have been revised as described in the response above and as shown in the markups provided with this response.

The large volume of water in the SFP prevents an undetected addition of unborated water sufficient to dilute the boron concentration from the minimum required concentration required by technical specifications to below the 800 ppm credited in the criticality analysis in Reference 9.1.1-1. The amount of water needed would be detected by pool level alarms and observed by operators. Once detected, the flow of unborated water would be stopped and the boron addition system could add borated water to the SFP to return the boron concentration to normal. Sufficient time is available for action to preclude a boron dilution event.

9.1.1.3.1 Analysis Code and Validation

The criticality analysis in Reference 9.1.1-1 uses the KENO-Va component of SCALE Version 6.1.3 (Reference 9.1.1-2) to calculate k_{eff} of the fuel in the storage racks. The calculations benchmark the ability of the criticality code to predict the reactivity of a system based on comparison to critical experiments. The criticality benchmark calculations establish the values of the calculational bias associated with the calculation methodology compared to benchmarks, and the standard deviation of the calculation bias. The validation determines the calculational bias and the uncertainty of the bias associated with the modeling methodology, the code, and the cross-section library for the criticality analysis. The validation uses the guidance of NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," (Reference 9.1.1-3) and NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," (Reference 9.1.1-4).

9.1.1.3.2 Analysis Conditions

The criticality analysis in Reference 9.1.1-1 incorporates the following assumptions for the fuel storage racks:

- maximum fuel assembly enrichment of 5 percent U-235 by weight, and no credit for burnup of the spent fuel assemblies
- no burnable neutron poisons or axial blankets present in the fuel assemblies
- fuel storage racks contain a new fuel assembly with the maximum reactivity in every storage location (even the locations in the racks not accessible by the fuel handling machine)

The analysis considers the following conditions to determine the maximum k_{eff} at a 95 percent probability and a 95 percent confidence level:

- moderator density variations with temperature to determine the optimum value
- manufacturing tolerances for the fuel storage racks and fuel assemblies for dimensions and material compositions
- fuel assembly storage positions within the fuel storage compartments, either the assemblies centered, or eccentric within the storage locations

RAI 09.01.01-1

- fuel cladding damage for five spent fuel assemblies

CP-0026

RAI 09.01.01-1

- abnormal conditions with ~~fuel cladding damage for five spent fuel assemblies,~~ the drop of a fuel assembly onto a storage rack, the drop of an assembly outside of the storage racks, and the transient deflection of the storage racks during a seismic event

CP-0026

9.1.1.3.3 Criticality Analysis Results

The maximum k_{eff} remains below the applicable limits in the regulations and no abnormal condition would cause an inadvertent criticality (Reference 9.1.1-1).

9.1.1.4 References

9.1.1-1 NuScale Power, LLC, "Fuel Storage Rack Analysis," TR-0816-49833, Revision 0.

9.1.1-2 Oak Ridge National Laboratory, "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," ORNL/TM-2005/39, Version 6.1, Oak Ridge, Tennessee, June 2011.

9.1.1-3 J.C. Dean and R.W. Tayloe, Jr, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” NUREG/CR-6698, Science Applications International Corporation, U.S. Nuclear Regulatory Commission, January 2001.

9.1.1-4 J. J. Lichtenwalter, S. M. Bowman, M. D. DeHart, and C. M. Hopper, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” NUREG/CR-6361 (ORNL/TM-13211), Oak Ridge National Laboratory, U.S. Nuclear Regulatory Commission, March 1997.

9.1.1-5 American National Standards Institute, “Design Requirements for Light Water Reactor Fuel Handling Systems,” ANSI/ANS-57.1-1992, La Grange Park, IL.

9.1.1-6 American National Standards Institute, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” ANSI/ANS-57.2-1983, La Grange Park, IL.

9.1.1-7 American National Standards Institute, “Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants,” ANSI/ANS-57.3-1983, La Grange Park, IL.

{{

}}2(a),(c)

3.3.6.2 Damaged Fuel Assembly

The fuel storage is required to hold up to five damaged FAs. For this analysis a damaged FA is assumed to have a cladding failure in 100 percent of the fuel rods where the gap between the fuel pellet and the clad is flooded with water.

The damaged fuel analysis is run with the whole pool model as shown in Figure 3—202, less the extra assembly shown for the drop analysis. Three scenarios with damaged FAs are simulated, as described below. The ~~analysis is run at~~analyses are run with both an unborated moderator and with a dissolved boron concentration of 800 ppm.

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- Damaged fuel is concentrated at the center of a fuel storage rack.

- Damaged fuel is positioned in corner locations within the~~near the corner of a~~ fuel storage rack where the poison plates are slightly narrower.

- Damaged fuel is concentrated around a single corner of a single rack.

- All FAs are assumed to be damaged.

The results are shown in Table 3-73. The values of $k_{95/95}$ shown in this table are calculated using Equation 1 in Section 3.3.1, using the appropriate values shown in Table 3-74.

Table 3-73 Fuel storage damaged fuel analysis

~~Scenario~~	~~k_{eff}~~	~~σ~~	~~$k_{95/95}$~~
~~No damaged fuel, full pool~~	~~0.86580~~	~~0.00030~~	~~0.89845~~
~~5 damaged FAs in center of pool~~	~~0.86650~~	~~0.00028~~	~~0.89915~~
~~5 damaged FAs near corner poison plates~~	~~0.86513~~	~~0.00030~~	~~0.89778~~
~~All damaged FAs~~	~~0.87337~~	~~0.00027~~	~~0.90602~~

Scenario	k_{eff}	σ	$k_{95/95}$	Limit
No damaged fuel, full pool, 800 ppm boron	0.86693	0.00009	0.90189	0.95
5 damaged FAs, in center of rack, 800 ppm boron	0.86719	0.00009	0.90215	0.95
5 damaged FAs, in corner locations, 800 ppm boron	0.86696	0.00009	0.90192	0.95
5 damaged FAs, in one corner, 800 ppm boron	0.86677	0.00009	0.90173	0.95
All damaged FAs, 800 ppm boron	0.87381	0.00009	0.90877	0.95
No damaged fuel, full pool, fresh water, 0 ppm boron	0.91602	0.00010	0.94630	1
5 damaged FAs, in center of rack, 0 ppm boron	0.91637	0.00010	0.94665	1
5 damaged FAs, in corner locations, 0 ppm boron	0.91590	0.00009	0.94618	1
5 damaged FAs, in one corner, 0 ppm boron	0.91592	0.00010	0.94620	1
All damaged FAs, 0 ppm boron	0.92407	0.00010	0.95434	1

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-2

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. 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 states that the reviewer should evaluate the normal- and abnormal- conditions models to verify that normal and abnormal conditions are modeled correctly and that all modeling approximations and assumptions, including omitted materials, are appropriate.

Points 5 and 6 on Page 290 of TR-0816-49833-P state that the spent fuel storage rack criticality model omits some rack structural components and materials beyond the active fuel length because replacing the effectively non- interacting material with a moderator would increase reactivity. While the staff agrees this may be true for non- borated moderator, the applicant credits soluble boron for abnormal conditions. Because boron is a neutron absorber, replacing rack components with borated moderator may be non-conservative.

Therefore, to demonstrate compliance with GDC 62 and 10 CFR 50.68, please provide justification that replacing structural components and materials of the fuel racks with borated water would be conservative (the moderation effect of the borated moderator outweighs the absorption effect) or would have a negligible effect, and update TR- 0816-49833-P with such justification; or, alternatively, update the analysis to include the currently omitted structural components and materials.

NuScale Response:

A sensitivity study was performed to consider the structural elements that were excluded from the original analysis. This included structural materials above the active fuel height and those within the active fuel height. The studies were performed with both a borated and unborated moderator.

Separate studies were performed for the elements located above and below the active fuel and for the support tubes located in the flux trap region. It was concluded that modeling the structural material above and below the fuel as an unborated reflector would provide an adequate representation. The results showed that there was a nearly statistical equal value when compared to the water only case. In the cases that examined the support bars or tubes in the flux trap region, the results showed that modeling the tubes would increase the reactivity by approximately 0.223% with an uncertainty of 0.0004 in an unborated or moderately borated (800 ppm boron) pool. From these studies, an additional bias and uncertainty in the bias of approximately 0.0022 +/- 0.0004 has been added to the criticality analyses. Applying this additional bias and uncertainty demonstrated that the results still meet the applicable limits. The affected tables and figures of TR-0816-49833 Rev. 0 have been revised accordingly as described below and shown in the attached markups.

Assumption 5 of Section 3.3.2 has been revised to identify that sensitivity studies were performed to determine the effect of the structural elements in the model and that from these studies, an additional bias and uncertainty in the bias is included in the determination of the $k_{95/95}$ values for the racks.

An additional paragraph has been added to Section 3.3.6.1 to describe the sensitivity cases for the structural elements.

Table 3-74 has been revised to include the bias and the uncertainty in the bias to account for the structural material evaluation. This table has also been updated based on the response to RAI Question 09.01.01-4.

In addition, the $k_{95/95}$ values in Tables 3-76 and 3-90, along with Figure 3-204, have been revised to account for the change in bias related to the responses to RAI Questions 09.01.01-2 and 09.01.01-4.

Impact on DCA:

Technical Report TR-0816-49833, Rev. 0, Fuel Storage Rack Analysis, Sections 3.3.2 and 3.3.6.1; Tables 3-74, 3-76, and 3-90; and Figure 3-204 have been revised as described in the response above and as shown in the markups provided with this response.

3. The fuel pellet density is not reduced by either the chamfer and or the dish. This conservatively increases the amount of fissionable material in the model.

4. The U-235 enrichment is assumed to be 5 wt percent, the maximum allowed by 10 CFR 50.68 (b)(7). Enrichment is not included in the tolerance analysis.

5. Sensitivity studies were performed to determine the effect of the structural elements in the model. From these studies, an additional bias and uncertainty in the bias of 0.0022 +/- 0.0004 is included in the determination of the $k_{95/95}$ values for the racks.~~Some rack structural components are omitted from the model, i.e. fuel storage rack spacer grids. This is conservative in that a material that acts as a neutron absorber is replaced by a material that acts as a neutron moderator. Replacing the effectively non-interacting material with a moderator would increase the reactivity.~~

6. Only the active fuel length is modeled, with a water reflector at the axial boundary. This is conservative in that FA structure and rack structure, including some poison plate length, is replaced by moderator. Because the amount of poison plate that is omitted is much larger than the tolerance on the length of the poison plate, the poison plate length is not included in the tolerance analysis.

7. The radial boundary conditions for the fuel storage single rack cases include a water gap that is governed by the dimensions of the base plate that supports the rack, with a periodic boundary condition that simulates an infinite array of fuel storage racks.

8. The radial boundary conditions for the fuel storage whole-pool cases include a water gap that is governed by the nominal distance to the pool wall, with a concrete reflector consistent with the pool wall.

9. The whole-pool cases for the fuel storage assume all rack locations are used, even though some locations are physically inaccessible.

3.3.3 Configuration

3.3.3.1 Fuel Storage Rack Model

The rack for the fuel storage is modeled in KENO-V.a with the nominal FA dimensions shown in Table 3-66 and the nominal rack dimensions shown in Table 3-67. The KENO-V.a model of a single rack location for the fuel storage is shown in Figure 3—200. The KENO-V.a model of a single fuel storage rack is shown in Figure 3—201.

The fuel storage rack is modeled with a radial reflector region that is half the minimum rack-to-rack spacing, with periodic radial boundary conditions that simulate an infinite array of racks. The regions directly above and below the fuel region are modeled as 13 inches of water followed by a water boundary condition.

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3.3.5 Software Use and Qualification

KENO-V.a controlled by the CSAS5 module of SCALE Version 6.1.3 is qualified for use. SCALE Version 6.1.3 is the software used to perform safety-related calculations in accordance with AREVA procedures.

3.3.6 Analysis, Evaluation, and Data

3.3.6.1 Fuel Storage Tolerance Analysis

The fuel storage tolerance analysis for both the fuel and the rack shown in Table 3-71 is performed with unborated moderator and the tolerance analysis shown in Table 3-72 is performed with the moderator at 800 ppm dissolved boron. The poison plate thickness does not affect the areal density of B-10.

The fuel storage tolerance analysis shows there is no significant trend of k-effective versus poison plate motion in the direction parallel to the face of the plate.

The flux trap cases for the fuel storage tolerance analysis are shown in Table 3-71 and Table 3-72. The delta of {{ }}2(a),(c) for the flux trap is determined by the tolerance on the thickness of the {{ }}2(a),(c) square tube that defines the minimum water gap thickness. This value is still larger than the spacing restrictions imposed by the grid assemblies at the top and bottom of the rack.

The FA position tolerance analysis for the fuel storage consists of a single case where all assemblies move diagonally to the common corner of each group of four locations.

The last rows of both Table 3-71 and Table 3-72 show the value of Δktol and σtol, for the respective boron concentration in the moderator. Δktol is calculated by summing all the positive values of Δk, while σtol is calculated as the statistical propagation (SRSS) of the respective values of ~~σk~~$\sigma_{\Delta k}$, where the corresponding value of Δk is positive.

Separate sensitivity studies were performed to assess the effect of the rack internal structural elements that include the elements located above and below the active fuel and the support tubes located in the flux trap region. The studies showed that modeling the structural material above and below the fuel as an unborated reflector would provide an adequate representation of those components. The results also showed that there was a nearly statistical equal value when compared to the water only case. In the cases that examined the support bars or tubes in the flux trap region, the results showed that modeling the tubes would increase the reactivity by approximately 0.223% with an uncertainty of 0.0004 in an unborated or moderately borated (800 ppm boron) pool. From these studies, the additional bias and uncertainty in the bias is 0.0022 +/- 0.0004 for 800 ppm boron, and 0.00223 +/- 0.00038 for no boron.

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Table 3-74 Factors for $k_{95/95}$ calculation for fuel storage

~~**Symbol**~~	~~**Value**~~		~~**Source**~~
~~Δk_{sys}~~	~~B @ 0 ppm~~	~~0.00420~~	~~Section 3.3.4.1~~
	~~B @ 800 ppm~~	~~0.00376~~	
~~$bias_m$~~	~~-0.00064~~		~~Section 3.3.7~~
~~C~~	~~1.993~~		~~Section 3.3.7~~
~~σ_{sys}~~	~~B @ 0 ppm~~	~~0.000382~~	~~Section 3.3.4.1~~
	~~B @ 800 ppm~~	~~0.000396~~	
~~σ_{bias}~~	~~0.00676~~		~~Section 3.3.7~~
~~**Tolerance Factors for 800 ppm Dissolved Boron**~~			
~~σ_{tol}~~	~~0.001596~~		~~Table 3-72~~
~~Δk_{tol}~~	~~0.02476~~		~~Table 3-72~~
~~**Tolerance Factor for Unborated Moderator**~~			
~~σ_{tol}~~	~~0.001145~~		~~Table 3-71~~
~~Δk_{tol}~~	~~0.01886~~		~~Table 3-71~~

Symbol	**Description**	**Value**	**Source**
$bias_m$	Bias associated with the calculation methodology compared to benchmark	-0.00064	Section 3.3.7
σ_{Bias}	Sigma, Experiment bias	0.00676	Section 3.3.7
C	C, confidence factor	1.99300	Section 3.3.7
Δk_{tol}	Manufacturing tolerance, no boron	0.01886	Table 3-71
σ_{tol}	Sigma, manufacturing tolerance, no boron	0.00108	Table 3-71
Δk_{tol}	Manufacturing tolerance, 800 ppm	0.02476	Table 3-72
σ_{tol}	Sigma, manufacturing tolerance, 800 ppm	0.00160	Table 3-72
Δk_{sys}	Operational bias for temperature, no boron	0.00404	Section 3.3.4.1
σ_{sys}	Sigma, operational bias for temperature, no boron	0.00144	Section 3.3.4.1

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Symbol	Description	Value	Source
Δk_{sys}	Operational bias for temperature, 800 ppm	0.00370	Section 3.3.4.1
σ_{sys}	Sigma, operational bias for temperature, 800 ppm	0.00157	Section 3.3.4.1
Δk_{sys}	Operational bias for structural material, no boron	0.00223	Section 3.3.6.1
σ_{sys}	Sigma, operational bias for structural material, no boron	0.00038	Section 3.3.6.1
Δk_{sys}	Operational bias for structural material, 800 ppm	0.0022	Section 3.3.6.1
σ_{sys}	Sigma, operational bias for structural material, 800 ppm	0.0004	Section 3.3.6.1

3.3.6.3 Assembly Dropped on Top of Rack

A dropped assembly lying on top of the rack in a horizontal position the fuel storage is at least {{ }}$^{2(a),(c)}$ from the top of the active fuel region of the assemblies stored in the rack.

For the fuel storage in water this distance represents approximately {{ }}$^{2(a),(c)}$, which is more than sufficient to neutronically uncouple the dropped assembly from the assemblies stored in the rack. Thus, an assembly dropped on top of a fuel storage rack in the SFP is not a criticality concern.

3.3.6.4 Assembly Dropped Outside of the Fuel Storage Rack

The SFP consists of 14 individual racks, arranged in three rows of five racks, with one corner rack omitted to allow for the FA elevator. For the nominal configuration, there is not enough space to drop a FA between two racks. The only location where an assembly can be dropped adjacent to a rack is in the empty corner with the elevator. In this scenario, the dropped FA is placed as close to the surrounding filled storage racks as possible. ~~This scenario is shown in~~ Figure 3—202 provides a general representation of the relative position for the dropped fuel assembly. Several cases were analyzed that considered a dropped FA including located in the corner near the three racks, as well as directly adjacent (face-to-face) and half-way between two adjacent fuel assemblies in a rack. The cases also spanned the full width of the rack. The location of the dropped assembly was shown to be statistically insignificant. The results are shown in Table 3-75. The values of $k_{95/95}$ shown in this table are calculated using Equation 1 in Section 3.3.1, using the appropriate values shown in Table 3-74. Thus, an assembly dropped outside of a fuel storage rack in the SFP is not a criticality concern.

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Table 3-76 k-effective and $k_{95/95}$ for a seismic event in the fuel storage

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}}2(a),(c)

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}}2(a),(c)

Figure 3—204 $k_{95/95}$ for a seismic event in the fuel storage

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3.3.8 Summary of Criticality Evaluations

The results of the criticality analysis are summarized as follows:

- The normal and accident condition temperatures in the spent fuel pool range from 40 degrees F to 212 degrees F. The criticality analysis is performed at a temperature of 67 degrees F, and the $k_{95/95}$ calculated including additional bias and uncertainty for a minimum temperature of at 40 degrees F.
- The maximum $k_{95/95}$ with full density moderation in unborated water remains below the applicable limit of 1.0 for fuel stored in the storage racks without credit for burnup.
- The maximum $k_{95/95}$ with full density moderation in 800 ppm (natural boron) borated water remains below the applicable limit of 0.95 for fuel stored in the storage racks without credit for burnup.
- No accident condition is identified that would cause an inadvertent criticality.
- Considered a range of gaps and the seismic analysis confirmed that the gap is still within the allowed range and no permanent deformation occurred.

A summary of the criticality analysis results are presented in Table 3-90.

Table 3-90 Summary of criticality analysis results

Conditions	$k_{95/95}$	Limit
Flooded with unborated water at full moderator density	0.94630~~0.94441~~	1.0
Flooded with water at 800 ppm boron, full moderator density	0.90193~~0.89941~~	0.95
Flooded with water at 800 ppm boron, full moderator density, dropped FA	0.90210~~0.89933~~	0.95
Flooded with water at 800 ppm boron, rack deformation of 1.140 cm compression per cell, which is ~~(~~greater than the seismic deformation of {{ }}$^{2(a),(c)}$~~0.010 in)~~	0.94157~~0.93926~~	0.95

The criticality analysis was completed without consideration for loading patterns or zoning consideration. Based on the conservative assumptions used in the criticality analysis, it is acceptable to place fuel in the racks in all cell locations.

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NuScale Power

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-3

SRP Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 instructs the reviewer to verify that abnormal conditions are modeled correctly and that all modeling approximations and assumptions are appropriate.

For the abnormal condition of an assembly dropped outside of the fuel storage rack, the staff observed that the current assumed location of the dropped fuel assembly, shown in Figure 3-202 of TR-0816-49833-P, does not appear to be the most limiting in terms of neutronic coupling. The staff expects that a fuel assembly dropped equidistant to and as close as possible to the three racks closest to the bottom left-hand corner of the figure would be most limiting.

Therefore, to demonstrate compliance with GDC 62 and 10 CFR 50.68, please either (a) provide justification as to why the current assumed location is conservative or (b) provide a new analysis of the more limiting location. In addition, update TR-0816-49833-P with this information accordingly.

NuScale Response:

In the original analyses, several locations were considered for the dropped fuel assembly.The variation in results was statistically insignificant. This would be expected as the relative contribution from the single dropped assembly in comparison to a full pool of fuel assemblies would be small. Additional cases were analyzed that considered a dropped fuel assembly including a fuel assembly located in the corner as described in the question, as well as directly adjacent (face-to-face) and half-way between two adjacent fuel assemblies in a rack. The cases also spanned the full width of the rack.The variation in the $k_{95/95}$ values ranged from 0.90141 (adjacent to row 1) to 0.90210 (adjacent to row 5). In reviewing the results, some corrections were also made for rack-to-rack and rack-to-dropped fuel assembly spacing. The results confirmed that the specific location of the dropped assembly is statistically insignificant.

Section 3.3.6.4 of TR-0816-49833 has been revised to clarify that Figure 3-202 is a general representation of a dropped fuel assembly. Figure 3-202 has been revised to show the case described in the question and Table 3-75 has been updated with the results.

Impact on DCA:

Technical Report TR-0816-49833, Rev. 0, Fuel Storage Rack Analysis, Section 3.3.6.4, Figure 3-202, and Table 3-75 have been revised as described in the response above and as shown in the markups provided with this response.

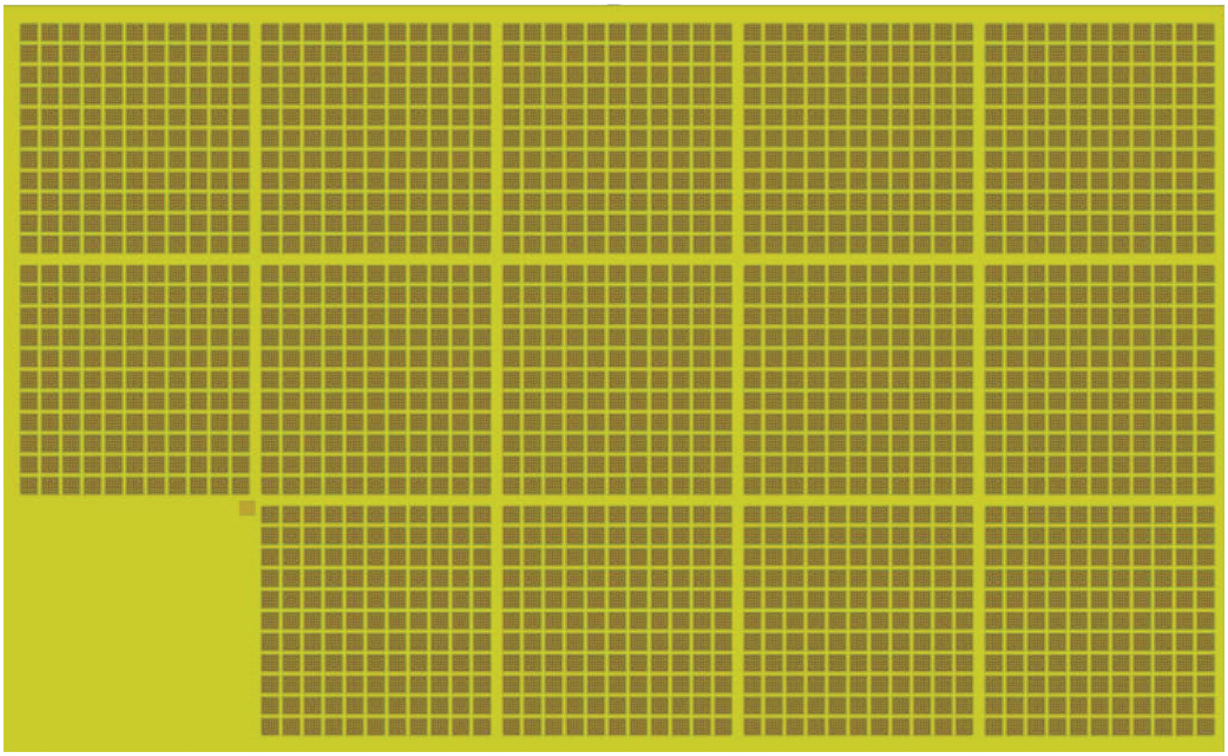

Figure 3—202 KENO-V.a model of the spent fuel pool

3.3.4 Initial Conditions, Boundary Conditions, and Limitations

3.3.4.1 Fuel Storage Model

The normal operating and accident condition temperature for the fuel storage racks is 40 degrees F to 212 degrees F (277.59 degrees K to 373.2 degrees K). The lowest available temperature in the 238 group ~~ENDV/B-VII~~ENDF/B-VII cross section library is 67 degrees F (292.59 degrees K). Therefore, to evaluate the sensitivity of the system to changes in temperature, cases are evaluated across a temperature range of 67 degrees F to 212 degrees F for 0 ppm boron and 800 ppm boron. The moderator density changes with the temperature for these cases, so the variation is due to both the temperature effect on the cross-section as well as the change in moderator density. The uncertainty terms for these cases are calculated using the root sum of squares of the base case and an uncertainty for the extrapolated value which was calculated by the statistical propagation of the errors of the curve fit coefficients.~~The moderator density is held at 1.0 g/cm^3 for these cases, so the variation is due to the temperature effect on the cross section. In order to evaluate the system for water temperatures as low as 40 degrees F, the reactivity trend calculated for the range of 67 degrees F to 212 degrees F is extrapolated to a temperature of 40 degrees F.~~ The results of the temperature sensitivity calculations and the extrapolated data points are provided in Table 3-69.

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Symbol	Description	Value	Source
Δk_{sys}	Operational bias for temperature, 800 ppm	0.00370	Section 3.3.4.1
σ_{sys}	Sigma, operational bias for temperature, 800 ppm	0.00157	Section 3.3.4.1
Δk_{sys}	Operational bias for structural material, no boron	0.00223	Section 3.3.6.1
σ_{sys}	Sigma, operational bias for structural material, no boron	0.00038	Section 3.3.6.1
Δk_{sys}	Operational bias for structural material, 800 ppm	0.0022	Section 3.3.6.1
σ_{sys}	Sigma, operational bias for structural material, 800 ppm	0.0004	Section 3.3.6.1

3.3.6.3 Assembly Dropped on Top of Rack

A dropped assembly lying on top of the rack in a horizontal position the fuel storage is at least {{ }}2(a),(c) from the top of the active fuel region of the assemblies stored in the rack.

For the fuel storage in water this distance represents approximately {{ }}2(a),(c), which is more than sufficient to neutronically uncouple the dropped assembly from the assemblies stored in the rack. Thus, an assembly dropped on top of a fuel storage rack in the SFP is not a criticality concern.

3.3.6.4 Assembly Dropped Outside of the Fuel Storage Rack

The SFP consists of 14 individual racks, arranged in three rows of five racks, with one corner rack omitted to allow for the FA elevator. For the nominal configuration, there is not enough space to drop a FA between two racks. The only location where an assembly can be dropped adjacent to a rack is in the empty corner with the elevator. In this scenario, the dropped FA is placed as close to the surrounding filled storage racks as possible. ~~This scenario is shown in~~ Figure 3—202 provides a general representation of the relative position for the dropped fuel assembly. Several cases were analyzed that considered a dropped FA including located in the corner near the three racks, as well as directly adjacent (face-to-face) and half-way between two adjacent fuel assemblies in a rack. The cases also spanned the full width of the rack. The location of the dropped assembly was shown to be statistically insignificant. The results are shown in Table 3-75. The values of $k_{95/95}$ shown in this table are calculated using Equation 1 in Section 3.3.1, using the appropriate values shown in Table 3-74. Thus, an assembly dropped outside of a fuel storage rack in the SFP is not a criticality concern.

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Table 3-75 Spent fuel pool dropped fuel assembly analysis

Scenario	k_{eff}	σ	$k_{95/95}$
Full pool, no dropped fuel	0.86697~~0.86684~~	0.00009~~0.00031~~	0.90193~~0.89949~~
1 dropped fuel assembly in fuel elevator area	0.86714~~0.86668~~	0.00014~~0.00026~~	0.90210~~0.89933~~

3.3.6.5 Misloaded Fuel Assembly

The fuel storage analysis conservatively assumes the racks are completely loaded with FAs at the maximum reactivity. There are no restrictions on loading patterns, therefore, there is no possibility of misloading an assembly in the fuel storage racks.

3.3.6.6 Fuel Storage Racks Seismic Event

The mechanical analysis of the fuel storage racks for a seismic event demonstrates that the racks do not undergo permanent deformation. However, there is a transient deflection of no more than ~~0.010 inch~~{{ }}2(a),(c) per storage tube (see Section 3.1.4.10.6). The seismic event is analyzed with the single fuel storage rack model and periodic radial boundary conditions. The spacing between storage tubes is reduced in a series of cases to determine the maximum deformation that maintains $k_{95/95}$ less than the limit of 0.95. The results are shown in Table 3-76 and Figure 3—204. The values of $k_{95/95}$ shown in this table are calculated using Equation 1 in Section 3.3.1, using the appropriate values shown in Table 3-74. {{

}}2(a),(c)

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-4

SRP Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 instructs 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. The reviewer is also to verify that normal-conditions models have been prepared for the full range of normal conditions.

In its criticality analyses, the applicant assumed a water density of 0.9982 g/cm^3, corresponding to a temperature of 67°F. According to Page 293 of TR-0816-49833-P, the spent fuel pool temperature range considering normal operation and accidents is 40-212°F. The staff notes that the water density assumed in the criticality analyses should correspond to the highest density possible within the spent fuel pool temperature range since this will maximize reactivity; in this case, the assumed density should correspond to that of 40°F water.

Therefore, to demonstrate compliance with GDC 62 and 10 CFR 50.68, please either (1) perform the criticality analyses assuming the maximum possible water density in the spent fuel pool or (2) provide justification for not assuming the maximum possible water density in the spent fuel pool. In addition, update TR-0816-49833-P accordingly.

NuScale Response:

Section 3.3.4.1 of TR-0816-49833 incorrectly states that the moderator density was held at 1.0 g/cm^3 for each of the cases used for the temperature extrapolation. The density was input at the correct value for each temperature. A bias term was developed based on an extrapolation to 40 degrees F to adjust to the density of 1.0 g/cm^3.

The following sections, tables, and figure of TR-0816-49833 have been revised for clarification.

- Section 3.3.1 has been revised to identify that a bias term, defined in Section 3.3.4.1, is included in the calculation of $k_{95/95}$ to adjust the system operating temperature to 40 degrees F and a density of 1.0 g/cm^3.
- Section 3.3.4.1 has been revised to clarify that the moderator density changes with the temperature, and so the variation of the system's response is due to both the temperature effect on the cross-section, as well as, the change in moderator density. Also, that the uncertainty terms are calculated using the root sum of squares of the base case and an uncertainty for the extrapolated value that was calculated by the statistical propagation of the errors of the curve fit coefficients.
- Table 3-67 has been updated to present the temperature range and rack array size information that is not proprietary.
- Table 3-69 has been updated with the revised results for operational tolerances.
- Figure 3-203 has been revised based on the changes to Table 3-69.
- Table 3-70 has been revised to include both the effect of temperature on the cross-sections and the effect of moderator density changes.
- Table 3-74 has been revised to clarify the operational bias for temperature.

Impact on DCA:

Technical Report TR-0816-49833, Rev. 0, Fuel Storage Rack Analysis, Sections 3.3.1 and 3.3.4.1; Tables 3-67, 3-69, 3-70, and 3-74; and Figure 3-203 have been revised as described in the response above and as shown in the markups provided with this response.

Criticality benchmark calculations are performed to establish the values of $bias_m$ and σ_{bias}. These calculations benchmark the ability of the criticality code to predict the reactivity of a system based on comparison to critical experiments. The criticality benchmark calculations and their applicability to the fuel rack analyses are documented in Section 3.3.7.

The fuel storage rack analyses assume the racks contain 17X17 fuel assemblies with a 0.374 in. fuel rod diameter and 0.496 in. fuel rod pitch. The FA parameters used for this analysis are shown in Table 3-66. The fuel storage rack analyses assume fresh fuel at the maximum allowable enrichment of 5 wt% U-235. The analyses do not take credit for burnup, zoning, or a loading pattern.

The design parameters for the racks that comprise the fuel storage are shown in Table 3-67. The material compositions are shown in Table 3-68. The analysis for the fuel storage uses a water density at 67 degrees F of 0.9982 g/cm^3. A bias term, defined in Section 3.3.4.1, is included in the calculation of $k_{95/95}$ to adjust the system operating temperature to 40 degrees F and a density of 1.0 g/cm^3.

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Table 3-67 Fuel rack design parameters for criticality analysis

~~{{~~

Description	Nominal Value	Tolerance(Note 2)
{{		
		}}2(a),(c)
Fuel storage rack array size	11x11	
Fuel storage temperature range	40 to 212°F	

{{

}}2(a),(c)

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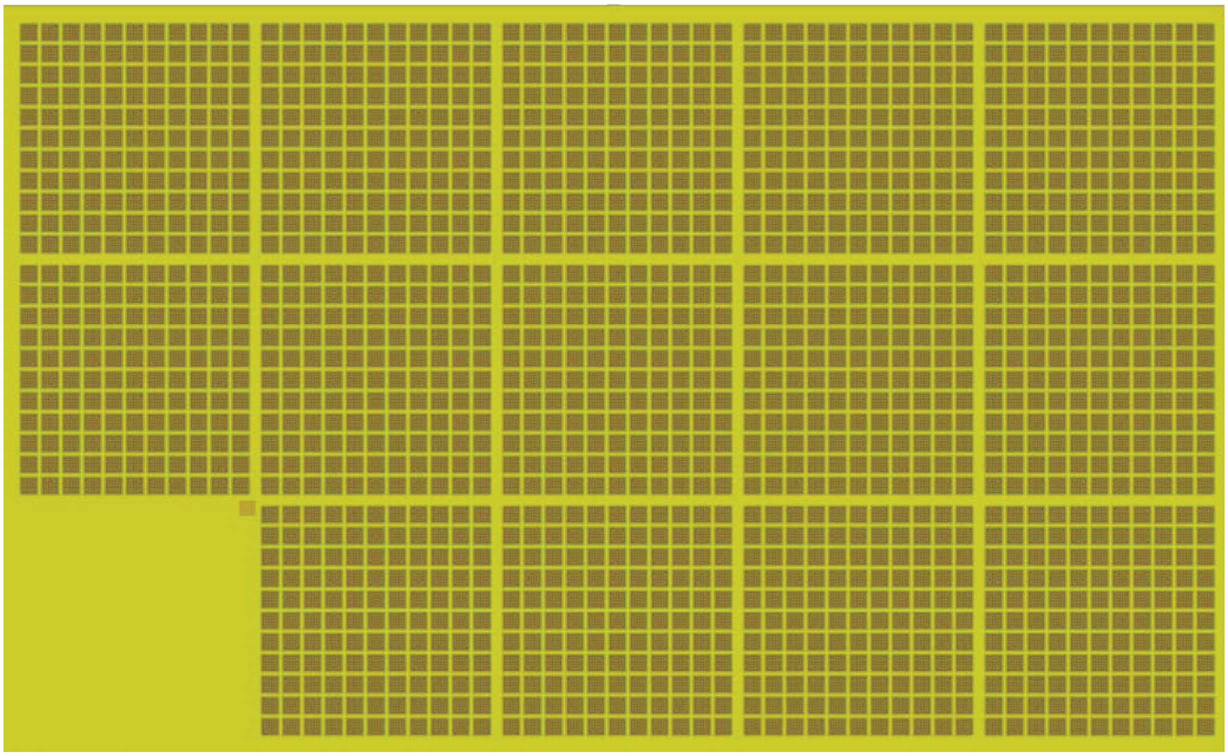

Figure 3—202 KENO-V.a model of the spent fuel pool

3.3.4 Initial Conditions, Boundary Conditions, and Limitations

3.3.4.1 Fuel Storage Model

The normal operating and accident condition temperature for the fuel storage racks is 40 degrees F to 212 degrees F (277.59 degrees K to 373.2 degrees K). The lowest available temperature in the 238 group ~~ENDV/B-VII~~ENDF/B-VII cross section library is 67 degrees F (292.59 degrees K). Therefore, to evaluate the sensitivity of the system to changes in temperature, cases are evaluated across a temperature range of 67 degrees F to 212 degrees F for 0 ppm boron and 800 ppm boron. The moderator density changes with the temperature for these cases, so the variation is due to both the temperature effect on the cross-section as well as the change in moderator density. The uncertainty terms for these cases are calculated using the root sum of squares of the base case and an uncertainty for the extrapolated value which was calculated by the statistical propagation of the errors of the curve fit coefficients.~~The moderator density is held at 1.0 g/cm³ for these cases, so the variation is due to the temperature effect on the cross section. In order to evaluate the system for water temperatures as low as 40 degrees F, the reactivity trend calculated for the range of 67 degrees F to 212 degrees F is extrapolated to a temperature of 40 degrees F.~~ The results of the temperature sensitivity calculations and the extrapolated data points are provided in Table 3-69.

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Figure 3—203 shows how the system responds, with a slightly negative gradient (-1x10^{-4} slope) to increases in temperature, with or without boron in the moderator.

Table 3-69 System sensitivity to nominal temperature

{{

}}2(a),(c)

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{{

}}2(a),(c)

Note: The 40 degrees F data points are extrapolated using the calculated data points and the line equation.

Figure 3—203 System sensitivity to nominal temperature

The operational tolerances for the 0 ppm and 800 ppm boron cases are calculated as the differential between the base (67 degrees F) case and the 40 degrees F extrapolated case. The uncertainty terms for these cases are calculated using the root sum of squares of the base case and an assumed uncertainty for the 40 degrees F extrapolated case of 0.00142~~0.00028~~. These operational tolerance values are applied in the $k_{95/95}$ calculation to account for the positive reactivity bias associated with the lower temperature of 40 degrees F.

Table 3-70 Operational tolerance impact on neutron multiplication

Conditions	**Δk_{sys}**	**$\sigma_{\Delta ksys}$**
Operational tolerance, 0 ppm	0.00404~~0.00420~~	0.00144~~0.000382~~
Operational tolerance, 800 ppm	0.00370~~0.00376~~	0.00157~~0.000396~~

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Table 3-74 Factors for $k_{95/95}$ calculation for fuel storage

~~**Symbol**~~	~~**Value**~~		~~**Source**~~
~~Δk_{sys}~~	~~B @ 0 ppm~~	~~0.00420~~	~~Section 3.3.4.1~~
	~~B @ 800 ppm~~	~~0.00376~~	
~~$bias_m$~~	~~-0.00064~~		~~Section 3.3.7~~
~~C~~	~~1.993~~		~~Section 3.3.7~~
~~σ_{sys}~~	~~B @ 0 ppm~~	~~0.000382~~	~~Section 3.3.4.1~~
	~~B @ 800 ppm~~	~~0.000396~~	
~~σ_{bias}~~	~~0.00676~~		~~Section 3.3.7~~
~~**Tolerance Factors for 800 ppm Dissolved Boron**~~			
~~σ_{tol}~~	~~0.001596~~		~~Table 3-72~~
~~Δk_{tol}~~	~~0.02476~~		~~Table 3-72~~
~~**Tolerance Factor for Unborated Moderator**~~			
~~σ_{tol}~~	~~0.001145~~		~~Table 3-71~~
~~Δk_{tol}~~	~~0.01886~~		~~Table 3-71~~

Symbol	**Description**	**Value**	**Source**
$bias_m$	Bias associated with the calculation methodology compared to benchmark	-0.00064	Section 3.3.7
σ_{Bias}	Sigma, Experiment bias	0.00676	Section 3.3.7
C	C, confidence factor	1.99300	Section 3.3.7
Δk_{tol}	Manufacturing tolerance, no boron	0.01886	Table 3-71
σ_{tol}	Sigma, manufacturing tolerance, no boron	0.00108	Table 3-71
Δk_{tol}	Manufacturing tolerance, 800 ppm	0.02476	Table 3-72
σ_{tol}	Sigma, manufacturing tolerance, 800 ppm	0.00160	Table 3-72
Δk_{sys}	Operational bias for temperature, no boron	0.00404	Section 3.3.4.1
σ_{sys}	Sigma, operational bias for temperature, no boron	0.00144	Section 3.3.4.1

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Symbol	Description	Value	Source
Δk_{sys}	Operational bias for temperature, 800 ppm	0.00370	Section 3.3.4.1
σ_{sys}	Sigma, operational bias for temperature, 800 ppm	0.00157	Section 3.3.4.1
Δk_{sys}	Operational bias for structural material, no boron	0.00223	Section 3.3.6.1
σ_{sys}	Sigma, operational bias for structural material, no boron	0.00038	Section 3.3.6.1
Δk_{sys}	Operational bias for structural material, 800 ppm	0.0022	Section 3.3.6.1
σ_{sys}	Sigma, operational bias for structural material, 800 ppm	0.0004	Section 3.3.6.1

3.3.6.3 Assembly Dropped on Top of Rack

A dropped assembly lying on top of the rack in a horizontal position the fuel storage is at least {{ }}$^{2(a),(c)}$ from the top of the active fuel region of the assemblies stored in the rack.

For the fuel storage in water this distance represents approximately {{ }}$^{2(a),(c)}$, which is more than sufficient to neutronically uncouple the dropped assembly from the assemblies stored in the rack. Thus, an assembly dropped on top of a fuel storage rack in the SFP is not a criticality concern.

3.3.6.4 Assembly Dropped Outside of the Fuel Storage Rack

The SFP consists of 14 individual racks, arranged in three rows of five racks, with one corner rack omitted to allow for the FA elevator. For the nominal configuration, there is not enough space to drop a FA between two racks. The only location where an assembly can be dropped adjacent to a rack is in the empty corner with the elevator. In this scenario, the dropped FA is placed as close to the surrounding filled storage racks as possible. ~~This scenario is shown in~~ Figure 3—202 provides a general representation of the relative position for the dropped fuel assembly. Several cases were analyzed that considered a dropped FA including located in the corner near the three racks, as well as directly adjacent (face-to-face) and half-way between two adjacent fuel assemblies in a rack. The cases also spanned the full width of the rack. The location of the dropped assembly was shown to be statistically insignificant. The results are shown in Table 3-75. The values of $k_{95/95}$ shown in this table are calculated using Equation 1 in Section 3.3.1, using the appropriate values shown in Table 3-74. Thus, an assembly dropped outside of a fuel storage rack in the SFP is not a criticality concern.

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-5

SRP Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 states that the configuration of spent fuel storage must prevent the insertion of a fuel assembly (FA) anywhere other than in a design location.

The applicant states in TR-0816-49833-P: "The design and size of the racks is set to provide a physical means to preclude misplacement of an FA between the wall and racks and between the racks." Similar statements are made in FSAR Tier 2, Subsections 9.1.1.2 and 9.1.2.2.2. However, from the illustration of the general arrangement of the racks in the spent fuel pool shown in Figure 1-2 of TR-0816-49833-P, it appears there is sufficient space between the racks and the pool wall (8 7/16", or 8.4375") to accommodate a fuel assembly (dimension = 8.426").

Therefore, to demonstrate compliance with GDC 62 and 10 CFR 50.68, please explain how the spacing between the racks and the pool wall precludes misplacement of a fuel assembly between the racks and the wall, and update TR- 0816-49833-P and FSAR Tier 2, Subsection 9.1.1.2 and 9.1.2.2.2 as necessary.

NuScale Response:

Misplacement of a fuel assembly between the racks and the wall is prevented by the design of the fuel handling machine and not by the spacing between the racks and the walls. The largest gap between the racks and a wall is slightly larger than the sides of a fuel assembly. But, as shown in Figure 1-2 of TR-0816-49833, the fuel handling machine has travel limitations. The red rectangle in the figure shows that the limited reach of the fuel handling machine prevents the insertion of a fuel assembly anywhere other than in a design location and cannot misplace a fuel assembly outside of a rack next to a wall.

Sections 9.1.1.2 and 9.1.2.2.2 of the FSAR, and Section 1.2 of TR-0816-49833 have been revised as shown in the attached to clarify that the fuel handling machine prevents misplacement of a fuel assembly outside of a rack near a pool wall.

Impact on DCA:

FSAR Sections 9.1.1.2 and 9.1.2.2.2; and Section 1.2 of Technical Report TR-0816-49833, Fuel Storage Rack Analysis, have been revised as described in the response above and as shown in the markups provided with this response.

CHAPTER 9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 Criticality Safety of Fresh and Spent Fuel Storage and Handling

9.1.1.1 Design Basis

This section identifies the required or credited functions for fresh and spent fuel storage and handling, the regulatory requirements that govern the performance of those functions, and the controlling parameters and associated values that ensure that the functions are fulfilled. Together, this information represents the design bases, defined in 10 CFR 50.2, as required by 10 CFR 52.47(a) and (a)(3)(ii).

General Design Criterion (GDC) 62, American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.1 (Reference 9.1.1-5), ANSI/ANS 57.2 (Reference 9.1.1-6), and ANSI/ANS 57.3 (Reference 9.1.1-7) were considered in the design of the storage and handling facility for new and spent fuel assemblies. Section 9.1.2 describes the protection of the fuel storage racks from natural phenomena.

The design and controls for operation of the fuel handling equipment and fuel storage racks prevent an inadvertent criticality using geometrically safe configurations, and using plant programs and procedures for criticality control. The fuel storage racks have an effective multiplication factor (k_{eff}) that meets 10 CFR 50.68.

9.1.1.2 Facilities Description

The storage and handling facility for new and spent fuel assemblies is located in the reactor building. The fuel storage racks in the spent fuel pool (SFP) can store either spent fuel assemblies, or new fuel assemblies. Section 9.1.2 describes the quantity of fuel assemblies that can be placed in the fuel storage racks and that travel limitations for the fuel handling machine prevent access to some of the fuel assembly storage locations.

RAI 09.01.01-5

The design of the fuel storage racks controls the center-to-center spacing between adjacent storage compartments for the fuel assemblies. The racks use square tubes for the fuel storage compartments. The spacing between compartments contains fixed neutron absorber plates and establishes flux traps. The neutron absorber plates use a boron carbide-aluminum metal matrix composite. The geometrically safe design of the fuel storage racks allows storage of new or spent fuel assemblies in any accessible location. The racks stand freely on the floor of the SFP. The layout in the SFP prevents an accidental placement of a fuel assembly between racks. The travel limitations for the fuel handling machine prevent misplacement of a fuel assembly ~~or~~ between a rack and a wall. As an abnormal condition, the criticality analysis in Reference 9.1.1-1 assumes a fuel assembly placed outside of, but next to, a rack in the corner of the SFP containing the new fuel elevator. The fuel assembly is assumed to be placed as close as possible to the filled storage racks.

Geometrically safe designs prevent criticality during fuel handling and for fuel stored in the fuel storage racks in accordance with 10 CFR 50.68(b). Fuel handling, described in Section 9.1.4, shows that the designs of the new fuel jib crane, the new fuel elevator, and the fuel handling machine allow each piece of equipment to move only a single fuel assembly at a time. Fuel handling procedures place controls on the movement of each fuel assembly and the designated storage location in a rack. The design of the fuel storage racks does not require loading patterns or zones for storage of the fuel assemblies. The design of the fuel storage racks eliminates the possibility of an inadvertent criticality occurring due to the selection of an inappropriate storage location or region. Plant programs and procedures track fuel assembly storage locations in accordance with special nuclear material regulations.

COL Item 9.1-1: A COL applicant that references the NuScale Power Plant design certification will develop plant programs and procedures for safe operations during handling and storage of new and spent fuel assemblies, including criticality control.

Section 9.1.2 describes the fuel storage racks; Section 9.1.3 describes the pool support systems; Section 9.1.4 describes the fuel handling equipment; and Section 9.1.5 describes heavy load handling.

9.1.1.3 Safety Evaluation

In accordance with GDC 62, the fuel handling equipment and fuel storage racks use geometrically safe configurations to prevent criticality. The design of the fuel handling equipment limits the number of new or spent fuel assemblies in motion to a single assembly for each piece of equipment. As described in Section 9.1.4, safety devices such as interlocks on the fuel handling equipment assist operators to prevent damage to an assembly and help minimize mishandling and movements not allowed by plant approved procedures.

The design of the fuel storage racks maintains stored fuel assemblies in subcritical arrays during normal and credible abnormal conditions. Reference 9.1.1-1 describes the details of the criticality analysis summarized below.

The fuel storage racks meet 10 CFR 50.68(b)(4) and have a k_{eff} no greater than 0.95, at a 95 percent probability and a 95 percent confidence level, for storage of new or spent fuel assemblies with credit for soluble boron. The analysis demonstrates that k_{eff} remains below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, for fuel stored in the racks with unborated water.

The large volume of water in the SFP prevents an undetected addition of unborated water sufficient to dilute the boron concentration from the minimum required concentration required by technical specifications to below the 800 ppm credited in moving spent fuel casks and NPMs. Section 9.4.2 describes the RXB heating and ventilation equipment that supports the environment above the SFP for personnel working in the area. Section 9.5.2 describes the communications systems, including the public address system. Section 9.1.3 describes the pool leakage detection system. Section 3.4.2 describes the design features to prevent groundwater intrusion into the below-ground portions of the RXB.

9.1.2.2.2 Fuel Storage Racks Design

RAI 09.01.01-5

There are 14 fuel storage racks and each has 121 storage locations in an 11 x 11 array for a total of 1,694 storage locations. As shown in Figure 9.1.2-1, travel limitations for the fuel handling machine prevent access to storage locations at the outer edges of the racks. The layout of the fuel storage racks in the SFP and the gaps between the fuel storage racks prevent the accidental placement of a fuel assembly between the fuel storage racks ~~or~~. The travel limitations for the fuel handling machine prevent misplacement of a fuel assembly between a fuel storage rack and ~~the~~a wall.

Each of the 14 fuel storage racks in the SFP is a free-standing structure. Reference 9.1.1-1 provides the general configuration and a cross-section of a fuel storage rack. Each consists of 121 square stainless steel tubes in a square array with neutron absorber plates and a space forming a flux trap located between each of the tubes. A single base plate supports the tubes. Braces and grid assemblies center the tubes and maintain spacing between the tubes. Legs under the baseplate provide space for entry of cooling water into a hole at the bottom of the storage compartments. Section 3.2 provides the safety and seismic classifications, and applicable quality assurance requirements, for the fuel storage racks.

Based on the travel limitations of the fuel handling machine, the fuel storage racks can safely store 1,404 fuel assemblies vertically in the SFP; however, only 1,393 fuel storage locations are considered accessible due to the possible difficulty reaching the storage locations closest to the weir wall. The 1,393 fuel storage locations include storage for five damaged fuel assemblies and for non-fuel core components such as a control rod assembly (stored within a fuel assembly).

The fuel handling machine cannot travel past the outer edge of the fuel storage racks except over the weir and into the corner of the SFP that has the new fuel elevator rather than a fuel storage rack. The possibility of a drop of a fuel assembly in this corner is not likely because of the single-failure proof design of the fuel handling machine and the use of safe handling procedures for criticality control. Section 9.1.1 addresses the unlikely event of a drop or placement of a fuel assembly in the corner containing the new fuel elevator.

The design of the fuel storage racks considers normal and postulated accident conditions including the effects of an SSE. The fuel storage racks meet Seismic Category I design requirements and perform their safety functions during and after an SSE. Reference 9.1.1-1 provides a description of the structural, thermal hydraulic, and criticality calculations for the fuel storage racks.

grids) and a larger gap between the racks and the pool walls (see Figure 1—2). {{ }}2(a),(c)

The fuel tubes are supported by a base plate with an area larger than the area of an 11x11 matrix of tubes, which is used to set the racks in appropriate proximity of each other (see Figure 1—3 and Figure 1—4). The racks are butted up against each other, which leaves no spacing between the racks, considering the baseplate and mid-level cross stiffener beam, but are not physically joined (see Figure 1—2 and Figure 1—3). The layout and gaps are made to prevent an accidental placement of an FA between racks~~ or between a rack and wall~~. The ~~design and size of the racks is set to provide a physical means to preclude~~travel limitations for the fuel handling machine prevent misplacement of an FA between ~~the~~a wall and a rack~~s and between the racks~~. Administrative controls are imposed to prevent placing FAs in the open space at the fuel elevator. Each of the rack modules is supported by {{ }}2(a),(c) independently adjustable feet to ensure level installation.

Standard material sizes are chosen to facilitate fabrication to the extent possible. The procurement and quality control procedures used in the fabrication are in accordance with the requirements of the ASME Code Section III, Subsection NF (Reference 12).

The installation of the racks in the pool may be performed in any sequence which best suits the erection contractor because there is no fuel present that would require special control of rack placement for criticality concerns. All feet of the rack are vertically adjustable and the racks are leveled prior to installation of the other racks. The feet are secured after the rack is level. The racks are butted up against each other and leave basically no spacing between the racks at the baseplate level. All racks are installed prior to filling water in the SFP.

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Docket No. 52-048

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Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-6

SRP Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 states that the normal- and abnormal-conditions models should address dimensional and material tolerances and uncertainties. In addition, the applicant cites ANSI/ANS-57.2, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” as guidance considered in the spent fuel storage facility design, and SRP Section 9.1.1 includes ANSI/ANS-57.2 as an acceptance criterion for meeting GDC ANSI/ANS-57.2 states that dimensional tolerances should be a factor in determining the fuel assembly arrangement with the highest value of k-effective.

The staff notes that the spent fuel pool tolerance analysis for unborated moderator in Table 3-71 of TR-0816- 49833-P does not appear to include the effects of an increase in fuel pellet outer diameter even though Table 3-72 for borated moderator includes it. For completeness of the spent fuel pool tolerance analysis for unborated moderator, and therefore compliance with GDC 62 and 10 CFR 50.68, please provide the results of considering an increase in fuel pellet outer diameter for the unborated moderator case, and update Table 3-71 of TR-0816-49833- P and any other calculations or conclusions that use the results of Table 3-71 accordingly.

NuScale Response:

Although Table 3-71 in TR-0816-49833-P, Rev 0, did not list an increase in fuel pellet diameter, this manufacturing tolerance is included in the analysis. The missing row has been added to the table and, as shown in the attached markup of TR-0816-49833-P, the last row of the table remains unchanged.

Impact on DCA:
Technical Report TR-0816-49833, Rev. 0, Fuel Storage Rack Analysis, Table 3-71 has been revised as described in the response above and as shown in the markups provided with this response.

Table 3-71 Spent fuel pool tolerance analysis with unborated moderator

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-7

SRP Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 states that the normal- and abnormal-conditions models should address dimensional and material tolerances and uncertainties. In addition, the applicant cites ANSI/ANS-57.2 as guidance considered in the spent fuel storage facility design, and SRP Section 9.1.1 includes ANSI/ANS-57.2 as an acceptance criterion for meeting GDC 62. ANSI/ANS-57.2 states that dimensional tolerances should be a factor in determining the fuel assembly arrangement with the highest value of k-effective.

Page 296 of TR-0816-49833-P states that the total uncertainty due to the tolerance analysis (σtol) is calculated as the square root of the sum of the squares (SRSS) of the individual uncertainty values for each tolerance examined. The staff notes that the values of σtol in Tables 3-71 and 3-72 of TR-0816-49833-P seem incorrect based its own confirmatory calculation and its understanding of the method to calculate σtot. Particularly, it appears that the reported values in Tables 3-71 and 3-72 under-calculate the uncertainty by 60 to 105 percent millirho. Because the uncertainty due to mechanical tolerances affects the margin to the criticality limits in 10 CFR 50.68, please confirm that the method used to calculate σtol is SRSS, as described in TR-0816-49833-P, and verify that the values for σtol were calculated correctly. If it is determined that errors were made in TR-0816-49833-P, update either (a) the calculation description or (b) Tables 3-71 and 3-72 and any other calculations or conclusions that use the information in Tables 3-71 and 3-72, as appropriate.

NuScale Response:

Page 296 of TR-0816-49833-P was incorrect in stating that the total uncertainty due to the tolerance analysis (σtol) is calculated as the square root of the sum of the squares (SRSS) of the individual uncertainty values for each tolerance examined. The calculation of σtol includes only the values of $\sigma_{\Delta k}$ where the corresponding value of Δk is positive. This is appropriate because the total Δk values at the bottom of Table 3-71 and Table 3-72 only include the positive values of Δk.

The value of σtol at the bottom of Table 3-71 inadvertently included the $\sigma_{\Delta k}$ for {{ }}$^{2(a)(c)}$, which had a negative Δk and should not have been included. As shown in the attached markup, the correct value of σtol for unborated water is {{ }}$^{2(a)(c)}$. This small conservative error has a minor effect on the values of $k_{95/95}$ provided later in the report. In addition, a correction was made to the values k_{eff}, σ, and Δk reported for the {{ }}$^{2(a)(c)}$ in the fresh water case in Table 3-71. The values were inadvertently taken from an incorrect case. Table 3-71 has been revised to use the values from the correct case as shown in the attached markups. No changes to Table 3-72 are needed.

The attached markups provide the following changes for TR-0816-49833:

- The text in the last paragraph of Section 3.3.6.1 has been revised to clarify that the calculation of σtol includes only the values of $\sigma_{\Delta k}$ where the corresponding value of Δk is positive.
- Table 3-71 has been updated as described above to not include the $\sigma_{\Delta k}$ for a negative value of Δk and to make a correction.

Impact on DCA:

Technical Report TR-0816-49833, Rev. 0, Fuel Storage Rack Analysis, Section 3.3.6.1 and Table 3-71 have been revised as described in the response above and as shown in the markups provided with this response.

3.3.5 Software Use and Qualification

KENO-V.a controlled by the CSAS5 module of SCALE Version 6.1.3 is qualified for use. SCALE Version 6.1.3 is the software used to perform safety-related calculations in accordance with AREVA procedures.

3.3.6 Analysis, Evaluation, and Data

3.3.6.1 Fuel Storage Tolerance Analysis

The fuel storage tolerance analysis for both the fuel and the rack shown in Table 3-71 is performed with unborated moderator and the tolerance analysis shown in Table 3-72 is performed with the moderator at 800 ppm dissolved boron. The poison plate thickness does not affect the areal density of B-10.

The fuel storage tolerance analysis shows there is no significant trend of k-effective versus poison plate motion in the direction parallel to the face of the plate.

The flux trap cases for the fuel storage tolerance analysis are shown in Table 3-71 and Table 3-72. The delta of {{ }}$^{2(a),(c)}$ for the flux trap is determined by the tolerance on the thickness of the {{ }}$^{2(a),(c)}$ square tube that defines the minimum water gap thickness. This value is still larger than the spacing restrictions imposed by the grid assemblies at the top and bottom of the rack.

The FA position tolerance analysis for the fuel storage consists of a single case where all assemblies move diagonally to the common corner of each group of four locations.

The last rows of both Table 3-71 and Table 3-72 show the value of Δktol and σtol, for the respective boron concentration in the moderator. Δktol is calculated by summing all the positive values of Δk, while σtol is calculated as the statistical propagation (SRSS) of the respective values of ~~σk~~$\sigma_{\Delta k}$, where the corresponding value of Δk is positive.

Separate sensitivity studies were performed to assess the effect of the rack internal structural elements that include the elements located above and below the active fuel and the support tubes located in the flux trap region. The studies showed that modeling the structural material above and below the fuel as an unborated reflector would provide an adequate representation of those components. The results also showed that there was a nearly statistical equal value when compared to the water only case. In the cases that examined the support bars or tubes in the flux trap region, the results showed that modeling the tubes would increase the reactivity by approximately 0.223% with an uncertainty of 0.0004 in an unborated or moderately borated (800 ppm boron) pool. From these studies, the additional bias and uncertainty in the bias is 0.0022 +/- 0.0004 for 800 ppm boron, and 0.00223 +/- 0.00038 for no boron.

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-8

SRP Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 instructs the reviewer to evaluate the computational method validation to verify that the validation study is thorough and uses benchmark critical experiments that are similar to the normal-conditions and credible abnormal-conditions models and to confirm that the k-effective bias and bias uncertainty values are properly determined.

In addition, the applicant cites NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," as guidance followed for the criticality code validation. NUREG/CR-6698 states that determining the bias and bias uncertainty in the calculation of k-effective involves determining a weighted mean that incorporates the uncertainty from both the measurement (σexp) and the calculation method (σcalc).

As described on Page 305 of TR-0816-49833-P, the applicant uses Equation 9 from NUREG/CR-6698, which normalizes the calculated k-effective if the critical experiment being modeled was at other than a critical state. Based on the values of σmeas presented in Table 3-86 of TR-0816-49833-P, the staff notes that the applicant did not use the correct experimental k-effective values for several critical experiment cases, particularly LEU- COMP-THERM-001, -002, -051, -070, and -075. In these cases, the applicant used the benchmark model k-effective rather than the experimental (actual measurement) k-effective.

Because the code bias and bias uncertainty affect the margin to the criticality limits in 10 CFR 50.68, please correct the values of σmeas for the identified cases and recalculate the code bias and bias uncertainty. Also, update any other calculations or conclusions in TR-0816-49833-P that use the code bias and bias uncertainty values.

NuScale Response:

The KENO models for the benchmark analysis for the cases noted in the RAI Question were taken from Reference 23 of TR-0816-49833 (shown below) with minor modifications to specify a newer neutron library. This reference evaluated the benchmark models that were provided, and for the cases noted in the RAI Question, determined that a minor modeling simplification would cause a small but non-negligible bias in the calculated results. This reference provided an adjusted value of the measured k-effective for each case that accounts for the simplification and eliminates this small bias. Note that each adjustment to the measured k-effective value was determined in the reference by an independent sensitivity study, and not by a comparison of the calculated k-effective to the measured k-effective.

Section 3.3.7.1.1 of TR-0816-49833 has been revised as shown in the attached markups to clarify that the measured k-effective is adjusted for some experiments where the published benchmark model has a known omission, which causes a small but non-negligible modeling bias.

Table 3-86 has been revised to add a footnote to clarify that the published benchmark model k_{eff} values were adjusted for experiments LEU-COMP-THERM-001, -002, -051, -070, and -075 (Reference 23) as described in Section 3.3.7.1.1.

Reference

23. Organization for Economic Co-operation and Development, Nuclear Energy Agency, “International Handbook of Evaluated Criticality Safety Benchmark Experiments,” NEA No. 7231, 2014.

Impact on DCA:

Section 3.3.7.1.1 and Table 3-86 of TR-0816-49833, Rev 0, have been revised as described in the response above and as shown in the markups provided with this response.

3.3.7.1 Methodology

3.3.7.1.1 Bias and Bias Uncertainty

When comparing the experimentally measured k-effective (k_{exp}) to the calculated k-effective (k_{calc}), the values are normalized as shown in NUREG/CR-6698~~6361~~, Equation 9 (Reference 20):

$$k_{\text{norm}} = {k_{\text{calc}}}/{k_{\text{exp}}}$$

The measured k-effective is adjusted for some experiments where the published benchmark model has a known omission, which causes a small but non-negligible modeling bias.

In addition, the errors are combined statistically as shown in NUREG/CR-6698~~6361~~, Equation 3~~9~~:

$$\sigma_t = \sqrt{\sigma_{\text{calc}}^2 + \sigma_{\text{exp}}^2}$$

The weighted mean value of k-effective is calculated with the following set of equations as shown in NUREG/CR-6698~~6361~~, Equations 4 – 7:

Variance about the mean:

$$s^2 = \frac{\left(\frac{1}{n-1}\right)\sum \frac{1}{\sigma_i^2}\left(k_{\text{norm}_i} - \overline{k_{\text{norm}}}\right)^2}{\frac{1}{n}\sum \frac{1}{\sigma_i^2}}$$

Average total uncertainty:

$$\overline{\sigma^2} = \frac{n}{\sum \frac{1}{\sigma_i^2}}$$

Weighted mean k_{norm}:

$$\overline{k_{\text{norm}}} = \frac{\sum \frac{1}{\sigma_i^2} k_{\text{norm}_i}}{\sum \frac{1}{\sigma_i^2}}$$

Square root of the pooled variance:

$$S_p = \sqrt{s^2 + \bar{\sigma}^2}$$

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3.3.7.3 Results of Benchmark Calculations

A summary of the pertinent parameters for each experiment is shown in Table 3-86 with the results of each KENO-V.a case.

Table 3-86 Critical experiment parameters and KENO-V.a results

#	Name	Enrichment (wt%)	Pitch (cm)	Separation (cm)	Boron (ppm)	Boron Plate (at/barn)	H/U	EALF (ev)	Benchmark Model (Note 1)		k-calc	σ-calc
									K_{eff}~~k-meas~~	σ_{exp}~~-meas~~		
1	LEU-COMP-THERM-001-001	2.35	2.032	0			2.918	9.67E-02	0.9998	0.0030	0.9980	0.00100
2	LEU-COMP-THERM-001-002	2.35	2.032	11.92			2.918	9.58E-02	0.9998	0.0030	0.9983	0.00110
3	LEU-COMP-THERM-001-003	2.35	2.032	8.41			2.918	9.50E-02	0.9998	0.0030	0.9983	0.00110
4	LEU-COMP-THERM-001-004	2.35	2.032	10.05			2.918	9.54E-02	0.9998	0.0030	0.9984	0.00095
5	LEU-COMP-THERM-001-005	2.35	2.032	6.39			2.918	9.43E-02	0.9998	0.0030	0.9959	0.00089
6	LEU-COMP-THERM-001-006	2.35	2.032	8.01			2.918	9.52E-02	0.9998	0.0030	0.9967	0.00100
7	LEU-COMP-THERM-001-007	2.35	2.032	4.46			2.918	9.36E-02	0.9998	0.0031	0.9972	0.00130
8	LEU-COMP-THERM-001-008	2.35	2.032	7.57			2.918	9.45E-02	0.9998	0.0031	0.9970	0.00096
9	LEU-COMP-THERM-002-001	4.306	2.54				3.882	1.13E-01	0.9997	0.0020	0.9976	0.00056
10	LEU-COMP-THERM-002-002	4.306	2.54				3.882	1.13E-01	0.9997	0.0020	0.9987	0.00029
11	LEU-COMP-THERM-002-003	4.306	2.54				3.882	1.13E-01	0.9997	0.0020	0.9983	0.00026
12	LEU-COMP-THERM-002-004	4.306	2.54				3.882	1.12E-01	0.9997	0.0018	0.9982	0.00028
13	LEU-COMP-THERM-002-005	4.306	2.54				3.882	1.10E-01	0.9997	0.0019	0.9968	0.00026
14	LEU-COMP-THERM-008-001	2.459	1.636		1511		1.841	2.80E-01	1.0007	0.0012	0.9975	0.00029
15	LEU-COMP-THERM-008-002	2.459	1.636		1334		1.841	2.47E-01	1.0007	0.0012	0.9985	0.00031
16	LEU-COMP-THERM-008-003	2.459	1.636		1337		1.841	2.47E-01	1.0007	0.0012	0.9983	0.00028
17	LEU-COMP-THERM-008-004	2.459	1.636		1183		1.841	2.46E-01	1.0007	0.0012	0.9978	0.00030
18	LEU-COMP-THERM-008-016	2.459	1.636		1158		1.841	2.28E-01	1.0007	0.0012	0.9973	0.00031
19	LEU-COMP-THERM-008-017	2.459	1.636		921		1.841	1.99E-01	1.0007	0.0012	0.9972	0.00037
20	LEU-COMP-THERM-009-001	4.306	2.54	8.58			3.882	1.13E-01	1.0000	0.0021	0.9988	0.00027
21	LEU-COMP-THERM-009-002	4.306	2.54	9.65			3.882	1.12E-01	1.0000	0.0021	0.9980	0.00029
22	LEU-COMP-THERM-009-003	4.306	2.54	9.22			3.882	1.13E-01	1.0000	0.0021	0.9981	0.00028
23	LEU-COMP-THERM-009-004	4.306	2.54	9.76			3.882	1.12E-01	1.0000	0.0021	0.9984	0.00028
24	LEU-COMP-THERM-009-009	4.306	2.54	6.72		7.92E-03	3.882	1.14E-01	1.0000	0.0021	0.9990	0.00025
25	LEU-COMP-THERM-010-005	4.306	2.54	14.255			3.882	3.56E-01	1.0000	0.0021	0.9997	0.00025
26	LEU-COMP-THERM-010-016	4.306	1.892	15.393			1.597	2.85E-01	1.0000	0.0028	1.0031	0.00031
27	LEU-COMP-THERM-010-017	4.306	1.892	15.363			1.597	2.79E-01	1.0000	0.0028	1.0016	0.00029
28	LEU-COMP-THERM-010-018	4.306	1.892	14.973			1.597	2.75E-01	1.0000	0.0028	1.0012	0.00030
29	LEU-COMP-THERM-010-019	4.306	1.892	13.343			1.597	2.68E-01	1.0000	0.0028	1.0013	0.00029
30	LEU-COMP-THERM-012-002	2.35	1.684 ~~2.032~~	3.86		9.20E-04		1.75E-01	1.0000	0.0034	0.9862	0.00088

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#	Name	Enrichment (wt%)	Pitch (cm)	Separation (cm)	Boron (ppm)	Boron Plate (at/barn)	H/U	EALF (ev)	Benchmark Model (Note 1)		k-calc	σ-calc
									K_{eff}~~k-meas~~	σ_{exp}~~-meas~~		
31	LEU-COMP-THERM-012-003	2.35	1.684 ~~2.032~~	3.46		1.40E-03		1.75E-01	1.0000	0.0034	0.9857	0.00093
32	LEU-COMP-THERM-012-004	2.35	1.684 ~~2.032~~	1.68		8.41E-03		1.83E-01	1.0000	0.0034	0.9890	0.00093
33	LEU-COMP-THERM-012-005	2.35	1.684 ~~2.032~~	1.93		8.73E-03		1.82E-01	1.0000	0.0034	0.9848	0.00091
34	LEU-COMP-THERM-013-001	4.306	1.892	13.273			1.597	2.86E-01	1.0000	0.0018	1.0003	0.00027
35	LEU-COMP-THERM-013-002	4.306	1.892	9.353		9.20E-04	1.597	2.94E-01	1.0000	0.0018	1.0000	0.00030
36	LEU-COMP-THERM-013-003	4.306	1.892	7.823		8.41E-03	1.597	2.98E-01	1.0000	0.0018	0.9995	0.00027
37	LEU-COMP-THERM-014-001	4.306	1.89		0		1.591	2.78E-01	1.0000	0.0019	0.9982	0.00027
38	LEU-COMP-THERM-014-002	4.306	1.89		490		1.591	3.33E-01	1.0000	0.0077	0.9861	0.00022
39	LEU-COMP-THERM-014-005	4.306	1.89		2550		1.591	5.84E-01	1.0000	0.0069	1.0007	0.00020
40	LEU-COMP-THERM-014-006	4.306	1.715		0		1.089	4.97E-01	1.0000	0.0033	1.0053	0.00023
41	LEU-COMP-THERM-014-007	4.306	1.715		1030		1.089	7.48E-01	1.0000	0.0051	1.0007	0.00022
42	LEU-COMP-THERM-016-012	2.35	2.032	6.33		7.92E-03		9.76E-02	1.0000	0.0031	0.9977	0.00110
43	LEU-COMP-THERM-016-013	2.35	2.032	9.03		7.92E-03		9.66E-02	1.0000	0.0031	0.9964	0.00140
44	LEU-COMP-THERM-016-014	2.35	2.032	5.05		7.92E-03		9.74E-02	1.0000	0.0031	0.9975	0.00084
45	LEU-COMP-THERM-039-001	4.7376	1.26				2	2.22E-01	1.0000	0.0014	0.9958	0.00030
46	LEU-COMP-THERM-039-002	4.7376	1.26				2.083	2.12E-01	1.0000	0.0014	0.9974	0.00034
47	LEU-COMP-THERM-039-003	4.7376	1.26				2.317	1.93E-01	1.0000	0.0014	0.9969	0.00074
48	LEU-COMP-THERM-039-004	4.7376	1.26				2.228	1.84E-01	1.0000	0.0014	0.9953	0.00085
49	LEU-COMP-THERM-039-005	4.7376	1.26				3.048	1.39E-01	1.0000	0.0009	0.9989	0.00083
50	LEU-COMP-THERM-039-006	4.7376	1.26				2.903	1.46E-01	1.0000	0.0009	0.9979	0.00095
51	LEU-COMP-THERM-039-007	4.7376	1.26				2	2.13E-01	1.0000	0.0012	0.9953	0.00095
52	LEU-COMP-THERM-039-008	4.7376	1.26				2.083	2.03E-01	1.0000	0.0012	0.9965	0.00090
53	LEU-COMP-THERM-039-009	4.7376	1.26				2.083	1.98E-01	1.0000	0.0012	0.9982	0.00085
54	LEU-COMP-THERM-039-010	4.7376	1.26				2.317	1.74E-01	1.0000	0.0012	0.9970	0.00085
55	LEU-COMP-THERM-039-011	4.7376	1.26				2	2.22E-01	1.0000	0.0013	0.9953	0.00079
56	LEU-COMP-THERM-039-012	4.7376	1.26				2	2.17E-01	1.0000	0.0013	0.9956	0.00086
57	LEU-COMP-THERM-039-013	4.7376	1.26				2	2.15E-01	1.0000	0.0013	0.9960	0.00083
58	LEU-COMP-THERM-039-014	4.7376	1.26				2	2.13E-01	1.0000	0.0013	0.9968	0.00093
59	LEU-COMP-THERM-039-015	4.7376	1.26				2	2.12E-01	1.0000	0.0013	0.9961	0.00085
60	LEU-COMP-THERM-039-016	4.7376	1.26				2	2.11E-01	1.0000	0.0013	0.9966	0.00085
61	LEU-COMP-THERM-039-017	4.7376	1.26				2	2.10E-01	1.0000	0.0013	0.9956	0.00089
62	LEU-COMP-THERM-042-003	2.35	1.684	2.69		8.41E-03		1.82E-01	1.0000	0.0016	0.9978	0.00054
63	LEU-COMP-THERM-048-001	3	1.32				0.996	6.76E-01	1.0000	0.0025	0.9985	0.00034
64	LEU-COMP-THERM-048-002	3	1.32				0.996	6.52E-01	1.0000	0.0025	0.9986	0.00028

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#	Name	Enrichment (wt%)	Pitch (cm)	Separation (cm)	Boron (ppm)	Boron Plate (at/barn)	H/U	EALF (ev)	Benchmark Model (Note 1)		k-calc	σ-calc
									K_{eff}~~k-meas~~	σ_{exp}~~-meas~~		
65	LEU-COMP-THERM-048-003	3	1.32				0.996	6.81E-01	1.0000	0.0025	0.9987	0.00024
66	LEU-COMP-THERM-048-004	3	1.32				0.996	6.83E-01	1.0000	0.0025	0.9982	0.00025
67	LEU-COMP-THERM-048-005	3	1.32				0.996	6.73E-01	1.0000	0.0025	0.9985	0.00026
68	LEU-COMP-THERM-050-001	4.738	1.3		0		2.203	2.00E-01	1.0000	0.0010	0.9981	0.00012
69	LEU-COMP-THERM-050-002	4.738	1.3		0		2.22	1.91E-01	1.0000	0.0010	0.9982	0.00012
70	LEU-COMP-THERM-050-003	4.738	1.3		822		2.189	2.07E-01	1.0000	0.0010	0.9986	0.00012
71	LEU-COMP-THERM-050-004	4.738	1.3		822		2.203	1.98E-01	1.0000	0.0010	0.9980	0.00011
72	LEU-COMP-THERM-050-005	4.738	1.3		5030		2.165	2.22E-01	1.0000	0.0010	0.9993	0.00012
73	LEU-COMP-THERM-050-006	4.738	1.3		5030		2.176	2.14E-01	1.0000	0.0010	0.9992	0.00012
74	LEU-COMP-THERM-050-007	4.738	1.3		5030		2.237	2.09E-01	1.0000	0.0010	0.9993	0.00012
75	LEU-COMP-THERM-051-010	2.459	1.636	1.636	15	4.82E-04		1.92E-01	1.0010	0.0019	0.9970	0.00011
76	LEU-COMP-THERM-051-011	2.459	1.636	1.636	28	4.84E-04		1.93E-01	1.0010	0.0019	0.9944	0.00013
77	LEU-COMP-THERM-051-012	2.459	1.636	1.636	92	3.75E-04		1.95E-01	1.0010	0.0019	0.9932	0.00011
78	LEU-COMP-THERM-051-013	2.459	1.636	1.636	395	1.20E-04		2.01E-01	1.0010	0.0022	0.9887	0.00011
79	LEU-COMP-THERM-051-014	2.459	1.636	3.272	121	1.20E-04		1.69E-01	1.0010	0.0019	0.9891	0.00011
80	LEU-COMP-THERM-051-015	2.459	1.636	1.636	487	7.23E-05		2.01E-01	1.0010	0.0024	0.9920	0.00011
81	LEU-COMP-THERM-051-016	2.459	1.636	3.272	197	7.23E-05		1.69E-01	1.0010	0.0020	0.9919	0.00012
82	LEU-COMP-THERM-051-017	2.459	1.636	1.636	634	2.99E-05		2.02E-01	1.0010	0.0027	0.9933	0.00012
83	LEU-COMP-THERM-051-018	2.459	1.636	3.272	320	2.99E-05		1.70E-01	1.0010	0.0021	0.9929	0.00011
84	LEU-COMP-THERM-051-019	2.459	1.636	4.908	72	2.99E-05		1.51E-01	1.0010	0.0019	0.9930	0.00011
85	LEU-COMP-THERM-070-001	6.5						1.49E+00	1.0004	0.0016	1.0055	0.00028
86	LEU-COMP-THERM-070-002	6.5						1.46E+00	1.0004	0.0016	1.0050	0.00031
87	LEU-COMP-THERM-070-003	6.5						1.45E+00	1.0004	0.0016	1.0048	0.00027
88	LEU-COMP-THERM-070-004	6.5						1.43E+00	1.0004	0.0016	1.0047	0.00028
89	LEU-COMP-THERM-070-005	6.5						1.40E+00	1.0004	0.0016	1.0045	0.00025
90	LEU-COMP-THERM-070-006	6.5						1.40E+00	1.0004	0.0016	1.0040	0.00027
91	LEU-COMP-THERM-070-007	6.5						1.39E+00	1.0004	0.0016	1.0039	0.00025
92	LEU-COMP-THERM-070-008	6.5						1.37E+00	1.0004	0.0016	1.0044	0.00027
93	LEU-COMP-THERM-070-009	6.5						1.36E+00	1.0004	0.0016	1.0038	0.00025
94	LEU-COMP-THERM-070-010	6.5						1.34E+00	1.0004	0.0016	1.0039	0.00026
95	LEU-COMP-THERM-070-011	6.5						1.32E+00	1.0004	0.0016	1.0032	0.00027
96	LEU-COMP-THERM-070-012	6.5						1.30E+00	1.0004	0.0016	1.0033	0.00027
97	LEU-COMP-THERM-075-004	6.5						1.45E+00	1.0004	0.0017	1.0061	0.00019
98	LEU-COMP-THERM-075-005	6.5						1.45E+00	1.0004	0.0017	1.0059	0.00020
99	LEU-COMP-THERM-075-006	6.5						1.45E+00	1.0004	0.0017	1.0063	0.00021

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#	Name	Enrichment (wt%)	Pitch (cm)	Separation (cm)	Boron (ppm)	Boron Plate (at/barn)	H/U	EALF (ev)	Benchmark Model (Note 1)		k-calc	σ-calc
									K_{eff} ~~k-meas~~	σ_{exp} ~~-meas~~		
100	PAT80SS1	4.74	1.6	2		6.44E-03	3.807	1.47E-01	1.0080	0.0140	1.0000	0.00190
101	PAT80SS2	4.74	1.6	2		6.44E-03	3.807	1.43E-01	1.0060	0.0140	0.9991	0.00170

Note 1: k_{eff} was adjusted for experiments LEU-COMP-THERM-001, -002, -051, -070, and -075 (Reference 23) as described in Section 3.3.7.1.1.

3.3.7.4 Trending Analysis

A regression analysis is performed to evaluate any potential bias that may trend with an independent variable. The following physical or spectral parameters are investigated:

- U-235 enrichment
- fuel rod pitch
- FA separation
- boron concentration in moderator
- boron areal density in separator plates
- moderator to fuel area ratio
- neutron spectrum as quantified by EALF

Each trend analysis uses a different set of experiments as described in Section 3.3.7.2.1 through Section 3.3.7.2.7. The results of the regression analysis are shown in Table 3-87.

The plots shown in Figure 3—205 through Figure 3—211 illustrate the data that the regression analysis used. The error bars plotted show the total uncertainty in the estimate, combining both the experimental and calculated uncertainty as shown in Table 3-86.

The regression analysis summarized in Table 3-87 shows that there is only a potential for three trending variables: assembly separation, soluble boron concentration, and areal boron density. These results trend with the highest correlation coefficient and passed the normality test (Shapiro-Wilk). Looking at the uncertainty in the fitting parameters and the magnitude of the slope, it can be seen that there is very little evidence of a measureable trend because the slope is so small whereas the uncertainty is relatively large. From this, it is clear there are no measureable trends present in the bias of calculations within the range of applicability using KENO-Va and the ENDF/B-VII, 238-group cross sections

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-9

SRP Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 62 and 10 CFR 50.68. SRP Section 9.1.1 instructs the reviewer to evaluate the computational method validation to verify that the validation study is thorough and uses benchmark critical experiments that are similar to the normal-conditions and credible abnormal-conditions models and to confirm that the k-effective bias and bias uncertainty values are properly determined, including the area of applicability for maximum k-effective values.

In addition, the applicant cites NUREG/CR-6698 as guidance followed for the criticality code validation. NUREG/CR-6698 describes an approach to develop the area of applicability of a system to be evaluated, including using the scope of selected criticality experiments to determine the detailed area of applicability that the experiments cover.

The staff notes that the area of applicability in Table 3-89 of TR-0816-49833-P does not appear consistent with what was provided in Table 3-86 for the last three parameters (boron areal density in separator plates, moderator to fuel ratio, and neutron spectrum). Because the area of applicability of the code bias and bias uncertainty defines how they can be applied, please either (1) justify the current numbers for the last three parameters in Table 3-89 or (2) update them to reflect the experiments in Table 3-86.

NuScale Response:

The range values in Table 3-89 of TR-0816-49833 for the bottom three entries were incorrect and have been revised to be consistent with Table 3-86.

The range of applicability for each parameter is defined by the range of the experiments used in the trend analysis in Section 3.3.7.4. The experiments used for the trend analysis for each parameter are provided in Sections 3.3.7.2.2 through 3.3.7.2.7, and the new Section 3.3.7.2.8. While the range of applicability was incorrectly reported for some parameters, the expected value for the fuel storage racks remains covered by the corrected ranges. While verifying the information, an error was identified in Table 3-86 in the fuel pin pitch data for

Experiment LEU-COMP-THERM-012 and it was corrected to 1.684 cm from 2.032 cm. The regression analysis for fuel pin pitch and EALF was recalculated.

The following sections, tables, and figures of TR-0816-49833 have been revised and a new section and table were added as follows:

- Section 3.3.7.2.4: The FA separation ranged from 0 cm to 15.393 cm, which was corrected from 1.68 cm to 15.393 cm. In addition, it should be noted that in Table 3-82, experiment LEU-COMP-THERM-048 was not used. Table 3-82 was corrected.
- Section 3.3.7.2.5: The soluble boron ranged from 0 ppm to 5030 ppm, which was corrected from 15 ppm to 5030 ppm.
- Section 3.3.7.2.6: The calculated boron atom density for the NuScale fuel storage racks is 2.8E-3 at/barn, which was corrected from 3.0E-4 at/barn.
- Section 3.3.7.2.7: The moderator to fuel ratios ranged from 0.966 to 3.882, which was corrected from 0.966 to 5.067.
- Section 3.3.7.2.8 was added to describe the experiments used for the EALF trend in Section 3.3.7.4. The EALF values ranges from 0.11 to 0.748 eV. Table 3-85a was added to list the experiments selected for EALF.
- Table 3-86 was corrected for the fuel pin pitch data for experiment LEU-COMP-THERM-012 as described above.
- Table 3-87 was corrected for the regression analysis for fuel pin pitch and for EALF.
- Figure 3-206 and Figure 3-211 were revised based on the new regression analysis for fuel pin pitch and EALF.
- Table 3-89 was revised as follows:
    - Boron areal density was changed to 3.0E-5 to 8.7E-3 at/barn, from 3.5E-5 to 8.7E-3 at/barn
    - Moderator to fuel ratio was changed to 0.996 to 3.882, from 0.996 to 5.067
    - EALF was changed to 0.11 to 0.748 eV, from 0.0198 to 0.748 eV
    - The estimated NuScale value for the boron areal density in separator plates was changed to 2.8E-3 from 4E-3 at/barn.

Impact on DCA:

Technical Report TR-0816-49833, Rev. 0, Fuel Storage Rack Analysis, new Section 3.3.7.2.8 and Table 3-85a were added. Also, Sections 3.3.7.2.4, 3.3.7.2.5, 3.3.7.2.6, and 3.3.7.2.7; Tables 3-82, 3-86, 3-87, and 3-89; and Figures 3-206 and 3-211 have been revised as described in the response above and as shown in the markups provided with this response.

Table 3-81 Benchmark experiments selected for fuel rod pitch trend

Experiment Name	Reference	Cases Selected	# of Cases
LEU-COMP-THERM-001	Ref. 23	1 – 8	8
LEU-COMP-THERM-002	Ref. 23	1 – 5	5
LEU-COMP-THERM-008	Ref. 23	1 – 4, 16, 17	6
LEU-COMP-THERM-009	Ref. 23	1 – 4	4
LEU-COMP-THERM-010	Ref. 23	5, 16 – 19	5
LEU-COMP-THERM-012	Ref. 23	2 – 5	4
LEU-COMP-THERM-013	Ref. 23	1, 2, 3	3
LEU-COMP-THERM-014	Ref. 23	1, 2, 5, 6, 7	5
LEU-COMP-THERM-016	Ref. 23	12 - 14	3
LEU-COMP-THERM-039	Ref. 23	1 – 17	17
LEU-COMP-THERM-042	Ref. 23	3	1
LEU-COMP-THERM-048	Ref. 23	1 – 5	5
LEU-COMP-THERM-050	Ref. 23	1 – 7	7
LEU-COMP-THERM-051	Ref. 23	10 – 19	10
PAT80	Ref. 21	SS1, SS2	2
Total Cases			**85**

3.3.7.2.4 Selection of Experiments for Fuel Assembly Separation Trend

The experiments shown in Table 3-82 are chosen to sample the calculation of k-effective with FA separation that is representative of the fuel storage racks. The FA separation ranged from ~~1.68~~zero cm to 15.393 cm. This adequately covers the expected FA separation for the fuel storage racks, which is approximately 7 cm. In order to obtain a wide range of assembly separation values, it is necessary to expand the area of applicability for enrichment to enrichments as low as 2.35 wt%. The result of expanding the area of applicability serves to improve the statistical significance, but did not change the results in any way.

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Table 3-82 Benchmark experiments selected for fuel assembly separation trend

Experiment Name	Reference	Cases Selected	# of Cases
LEU-COMP-THERM-001	Ref. 23	1 – 8	8
LEU-COMP-THERM-009	Ref. 23	1 – 4	4
LEU-COMP-THERM-010	Ref. 23	5, 16 – 19	5
LEU-COMP-THERM-012	Ref. 23	2 – 5	4
LEU-COMP-THERM-013	Ref. 23	1, 2, 3	3
LEU-COMP-THERM-016	Ref. 23	12, 13, 14	3
LEU-COMP-THERM-042	Ref. 23	3	1
~~LEU-COMP-THERM-048~~	~~Ref. 23~~	~~1 – 5~~	~~5~~
LEU-COMP-THERM-051	Ref. 23	10 – 19	10
PAT80	Ref. 21	SS1, SS2	2
Total Cases			**~~45~~40**

3.3.7.2.5 Selection of Experiments for Soluble Boron Trend

The experiments shown in Table 3-83 are chosen to sample the calculation of k-effective with soluble boron concentration that is representative of the fuel storage racks. The soluble boron concentration ranged from ~~15~~0 ppm to 5030 ppm. This adequately covers the expected soluble boron concentration and the modeled scenario of low-boron concentration for the fuel storage racks, of 2000 ppm and 200 ppm, respectively. In order to obtain a wider range of boron concentration values, it is necessary to expand the area of applicability for enrichment to enrichments as low as 2.35 wt%. Without this extension, there are only 12 experiments in the range of the nominal boron concentration and none in the range of the low-boron concentration used.

Table 3-83 Benchmark experiments selected for soluble boron trend

Experiment Name	Reference	Cases Selected	# of Cases
LEU-COMP-THERM-008	Ref. 23	1 – 4, 16, 17	6
LEU-COMP-THERM-014	Ref. 23	2, 5, 7	3
LEU-COMP-THERM-050	Ref. 23	2 – 7	6
LEU-COMP-THERM-051	Ref. 23	10 – 19	10
Total Cases			**25**

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3.3.7.2.6 Selection of Experiments for Boron Separator Plate Areal Density Trend

The experiments shown in Table 3-84 are chosen to sample the calculation of k-effective with boron separator plates that are representative of the NuScale fuel storage racks. The boron atom density ranged from 3E-5 to about 9E-3 at/barn. This adequately covers the expected boron density for the fuel storage racks, of approximately ~~3E-4~~2.8E-3 at/barn. In order to obtain a wider range of boron densities, it is necessary to expand the area of applicability for enrichment to enrichments as low as 2.35 wt%. Additionally, it is not possible to analyze the impact of boron areal density within the suggested ±20 percent, so the range of boron concentration in the separator plates is increased until a minimal sample size is obtained. Without this extension, there are only five experiments in the range of the nominal boron density, which is far too small a sample to determine if a trending bias is present. Also, there are Boroflex™ experiments that are not chosen, as this material is specifically excluded from the construction of the fuel storage racks. Cases that use stainless steel separator plates are included, even when impregnated with boron, because it is expected that the construction will make use of large amounts of stainless steel, if not for the poison plate substrate, it will still separate the FAs.

Table 3-84 Benchmark experiments selected for separator plate boron areal density trend

Experiment Name	Reference	Cases Selected	# of Cases
LEU-COMP-THERM-009	Ref. 23	9	1
LEU-COMP-THERM-012	Ref. 23	2 – 5	4
LEU-COMP-THERM-013	Ref. 23	2, 3	2
LEU-COMP-THERM-016	Ref. 23	12, 13, 14	3
LEU-COMP-THERM-042	Ref. 23	3	1
LEU-COMP-THERM-051	Ref. 23	10 – 19	10
PAT80	Ref. 21	SS1, SS2	2
Total Cases			**23**

3.3.7.2.7 Selection of Experiments for Moderator to Fuel Area Ratio Trend

The experiments shown in Table 3-85 are chosen to sample the calculation of k-effective with moderator-to-fuel ratios that are representative of the fuel storage racks. The range of ratios is from 0.996 to 3.882~~5.067~~. This adequately covers the expected ratio for the fuel storage racks of approximately 1.99. In most cases, the moderator-to-fuel ratio is not given in any reference, so a simple calculation is performed to compare the area of the fuel to the area of the moderator.

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Table 3-85 Benchmark experiments selected for moderator to fuel area ratio trend

Experiment Name	Reference	Cases Selected	# of Cases
LEU-COMP-THERM-002	Ref. 23	1 – 5	5
LEU-COMP-THERM-009	Ref. 23	1 – 4, 9	5
LEU-COMP-THERM-010	Ref. 23	5, 16 – 19	5
LEU-COMP-THERM-013	Ref. 23	1 – 3	3
LEU-COMP-THERM-014	Ref. 23	1, 2, 5 – 7	5
LEU-COMP-THERM-039	Ref. 23	1 – 17	17
LEU-COMP-THERM-048	Ref. 23	1 – 5	5
LEU-COMP-THERM-050	Ref. 23	1 – 7	7
Total Cases			**52**

3.3.7.2.8 Selection of Experiments for Neutron Spectrum

The experiments shown in Table 3-85a are chosen to sample the calculation of k_{eff} with the neutron spectrum, as represented by the Energy of Average Lethargy of Fission (EALF), that was representative of the NuScale fuel storage racks. The EALF ranged from 0.11 eV to 0.748 eV. This adequately covers the expected neutron spectrum of 0.3 eV.

Table 3-85a Benchmark experiments selected for neutron spectrum

Experiment Name	**Reference**	**Cases Selected**	**# of Cases**
LEU-COMP-THERM-002	Ref. 23	1-5	5
LEU-COMP-THERM-009	Ref. 23	1-4	4
LEU-COMP-THERM-010	Ref. 23	5,16-19	5
LEU-COMP-THERM-013	Ref. 23	1-3	3
LEU-COMP-THERM-014	Ref. 23	1,2,5,6,7	5
LEU-COMP-THERM-039	Ref. 23	1-17	17
LEU-COMP-THERM-048	Ref. 23	1-5	5
LEU-COMP-THERM-050	Ref. 23	1-7	7
PAT80	Ref. 21	SS1, SS2	2
Total Cases			53

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3.3.7.3 Results of Benchmark Calculations

A summary of the pertinent parameters for each experiment is shown in Table 3-86 with the results of each KENO-V.a case.

Table 3-86 Critical experiment parameters and KENO-V.a results

#	Name	Enrichment (wt%)	Pitch (cm)	Separation (cm)	Boron (ppm)	Boron Plate (at/barn)	H/U	EALF (ev)	Benchmark Model (Note 1)		k-calc	σ-calc
									K_{eff}~~k-meas~~	σ_{exp}~~-meas~~		
1	LEU-COMP-THERM-001-001	2.35	2.032	0			2.918	9.67E-02	0.9998	0.0030	0.9980	0.00100
2	LEU-COMP-THERM-001-002	2.35	2.032	11.92			2.918	9.58E-02	0.9998	0.0030	0.9983	0.00110
3	LEU-COMP-THERM-001-003	2.35	2.032	8.41			2.918	9.50E-02	0.9998	0.0030	0.9983	0.00110
4	LEU-COMP-THERM-001-004	2.35	2.032	10.05			2.918	9.54E-02	0.9998	0.0030	0.9984	0.00095
5	LEU-COMP-THERM-001-005	2.35	2.032	6.39			2.918	9.43E-02	0.9998	0.0030	0.9959	0.00089
6	LEU-COMP-THERM-001-006	2.35	2.032	8.01			2.918	9.52E-02	0.9998	0.0030	0.9967	0.00100
7	LEU-COMP-THERM-001-007	2.35	2.032	4.46			2.918	9.36E-02	0.9998	0.0031	0.9972	0.00130
8	LEU-COMP-THERM-001-008	2.35	2.032	7.57			2.918	9.45E-02	0.9998	0.0031	0.9970	0.00096
9	LEU-COMP-THERM-002-001	4.306	2.54				3.882	1.13E-01	0.9997	0.0020	0.9976	0.00056
10	LEU-COMP-THERM-002-002	4.306	2.54				3.882	1.13E-01	0.9997	0.0020	0.9987	0.00029
11	LEU-COMP-THERM-002-003	4.306	2.54				3.882	1.13E-01	0.9997	0.0020	0.9983	0.00026
12	LEU-COMP-THERM-002-004	4.306	2.54				3.882	1.12E-01	0.9997	0.0018	0.9982	0.00028
13	LEU-COMP-THERM-002-005	4.306	2.54				3.882	1.10E-01	0.9997	0.0019	0.9968	0.00026
14	LEU-COMP-THERM-008-001	2.459	1.636		1511		1.841	2.80E-01	1.0007	0.0012	0.9975	0.00029
15	LEU-COMP-THERM-008-002	2.459	1.636		1334		1.841	2.47E-01	1.0007	0.0012	0.9985	0.00031
16	LEU-COMP-THERM-008-003	2.459	1.636		1337		1.841	2.47E-01	1.0007	0.0012	0.9983	0.00028
17	LEU-COMP-THERM-008-004	2.459	1.636		1183		1.841	2.46E-01	1.0007	0.0012	0.9978	0.00030
18	LEU-COMP-THERM-008-016	2.459	1.636		1158		1.841	2.28E-01	1.0007	0.0012	0.9973	0.00031
19	LEU-COMP-THERM-008-017	2.459	1.636		921		1.841	1.99E-01	1.0007	0.0012	0.9972	0.00037
20	LEU-COMP-THERM-009-001	4.306	2.54	8.58			3.882	1.13E-01	1.0000	0.0021	0.9988	0.00027
21	LEU-COMP-THERM-009-002	4.306	2.54	9.65			3.882	1.12E-01	1.0000	0.0021	0.9980	0.00029
22	LEU-COMP-THERM-009-003	4.306	2.54	9.22			3.882	1.13E-01	1.0000	0.0021	0.9981	0.00028
23	LEU-COMP-THERM-009-004	4.306	2.54	9.76			3.882	1.12E-01	1.0000	0.0021	0.9984	0.00028
24	LEU-COMP-THERM-009-009	4.306	2.54	6.72		7.92E-03	3.882	1.14E-01	1.0000	0.0021	0.9990	0.00025
25	LEU-COMP-THERM-010-005	4.306	2.54	14.255			3.882	3.56E-01	1.0000	0.0021	0.9997	0.00025
26	LEU-COMP-THERM-010-016	4.306	1.892	15.393			1.597	2.85E-01	1.0000	0.0028	1.0031	0.00031
27	LEU-COMP-THERM-010-017	4.306	1.892	15.363			1.597	2.79E-01	1.0000	0.0028	1.0016	0.00029
28	LEU-COMP-THERM-010-018	4.306	1.892	14.973			1.597	2.75E-01	1.0000	0.0028	1.0012	0.00030
29	LEU-COMP-THERM-010-019	4.306	1.892	13.343			1.597	2.68E-01	1.0000	0.0028	1.0013	0.00029
30	LEU-COMP-THERM-012-002	2.35	1.684 ~~2.032~~	3.86		9.20E-04		1.75E-01	1.0000	0.0034	0.9862	0.00088

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#	Name	Enrichment (wt%)	Pitch (cm)	Separation (cm)	Boron (ppm)	Boron Plate (at/barn)	H/U	EALF (ev)	Benchmark Model (Note 1)		k-calc	σ-calc
									K_{eff} ~~k-meas~~	σ_{exp} ~~-meas~~		
31	LEU-COMP-THERM-012-003	2.35	1.684 ~~2.032~~	3.46		1.40E-03		1.75E-01	1.0000	0.0034	0.9857	0.00093
32	LEU-COMP-THERM-012-004	2.35	1.684 ~~2.032~~	1.68		8.41E-03		1.83E-01	1.0000	0.0034	0.9890	0.00093
33	LEU-COMP-THERM-012-005	2.35	1.684 ~~2.032~~	1.93		8.73E-03		1.82E-01	1.0000	0.0034	0.9848	0.00091
34	LEU-COMP-THERM-013-001	4.306	1.892	13.273			1.597	2.86E-01	1.0000	0.0018	1.0003	0.00027
35	LEU-COMP-THERM-013-002	4.306	1.892	9.353		9.20E-04	1.597	2.94E-01	1.0000	0.0018	1.0000	0.00030
36	LEU-COMP-THERM-013-003	4.306	1.892	7.823		8.41E-03	1.597	2.98E-01	1.0000	0.0018	0.9995	0.00027
37	LEU-COMP-THERM-014-001	4.306	1.89		0		1.591	2.78E-01	1.0000	0.0019	0.9982	0.00027
38	LEU-COMP-THERM-014-002	4.306	1.89		490		1.591	3.33E-01	1.0000	0.0077	0.9861	0.00022
39	LEU-COMP-THERM-014-005	4.306	1.89		2550		1.591	5.84E-01	1.0000	0.0069	1.0007	0.00020
40	LEU-COMP-THERM-014-006	4.306	1.715		0		1.089	4.97E-01	1.0000	0.0033	1.0053	0.00023
41	LEU-COMP-THERM-014-007	4.306	1.715		1030		1.089	7.48E-01	1.0000	0.0051	1.0007	0.00022
42	LEU-COMP-THERM-016-012	2.35	2.032	6.33		7.92E-03		9.76E-02	1.0000	0.0031	0.9977	0.00110
43	LEU-COMP-THERM-016-013	2.35	2.032	9.03		7.92E-03		9.66E-02	1.0000	0.0031	0.9964	0.00140
44	LEU-COMP-THERM-016-014	2.35	2.032	5.05		7.92E-03		9.74E-02	1.0000	0.0031	0.9975	0.00084
45	LEU-COMP-THERM-039-001	4.7376	1.26				2	2.22E-01	1.0000	0.0014	0.9958	0.00030
46	LEU-COMP-THERM-039-002	4.7376	1.26				2.083	2.12E-01	1.0000	0.0014	0.9974	0.00034
47	LEU-COMP-THERM-039-003	4.7376	1.26				2.317	1.93E-01	1.0000	0.0014	0.9969	0.00074
48	LEU-COMP-THERM-039-004	4.7376	1.26				2.228	1.84E-01	1.0000	0.0014	0.9953	0.00085
49	LEU-COMP-THERM-039-005	4.7376	1.26				3.048	1.39E-01	1.0000	0.0009	0.9989	0.00083
50	LEU-COMP-THERM-039-006	4.7376	1.26				2.903	1.46E-01	1.0000	0.0009	0.9979	0.00095
51	LEU-COMP-THERM-039-007	4.7376	1.26				2	2.13E-01	1.0000	0.0012	0.9953	0.00095
52	LEU-COMP-THERM-039-008	4.7376	1.26				2.083	2.03E-01	1.0000	0.0012	0.9965	0.00090
53	LEU-COMP-THERM-039-009	4.7376	1.26				2.083	1.98E-01	1.0000	0.0012	0.9982	0.00085
54	LEU-COMP-THERM-039-010	4.7376	1.26				2.317	1.74E-01	1.0000	0.0012	0.9970	0.00085
55	LEU-COMP-THERM-039-011	4.7376	1.26				2	2.22E-01	1.0000	0.0013	0.9953	0.00079
56	LEU-COMP-THERM-039-012	4.7376	1.26				2	2.17E-01	1.0000	0.0013	0.9956	0.00086
57	LEU-COMP-THERM-039-013	4.7376	1.26				2	2.15E-01	1.0000	0.0013	0.9960	0.00083
58	LEU-COMP-THERM-039-014	4.7376	1.26				2	2.13E-01	1.0000	0.0013	0.9968	0.00093
59	LEU-COMP-THERM-039-015	4.7376	1.26				2	2.12E-01	1.0000	0.0013	0.9961	0.00085
60	LEU-COMP-THERM-039-016	4.7376	1.26				2	2.11E-01	1.0000	0.0013	0.9966	0.00085
61	LEU-COMP-THERM-039-017	4.7376	1.26				2	2.10E-01	1.0000	0.0013	0.9956	0.00089
62	LEU-COMP-THERM-042-003	2.35	1.684	2.69		8.41E-03		1.82E-01	1.0000	0.0016	0.9978	0.00054
63	LEU-COMP-THERM-048-001	3	1.32				0.996	6.76E-01	1.0000	0.0025	0.9985	0.00034
64	LEU-COMP-THERM-048-002	3	1.32				0.996	6.52E-01	1.0000	0.0025	0.9986	0.00028

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within Version 6.1.3 of SCALE. Therefore, a single-sided tolerance factor is used to quantify the bias and uncertainty for the fuel storage racks.

Table 3-87 Regression analysis for possible bias trending variables

Parameter	Enrichment	Pitch	Separation	Soluble Boron	Plate Boron	Mod/ Fuel	EALF
Slope	2.1E-03 ±19 percent	~~4.0~~ 1.4E-04 ±>74~~100~~ percent	8E-04 ±13 percent	1.5E-06 ±31 percent	4.3E-01 ±56 percent	-2E-04 ±>100 percent	~~3.9~~ 4.2E-03 ±52~~54~~ percent
Intercept	0.9890	0.9942~~0.9959~~	0.9904	0.9933	0.9912	0.9986	0.9968 ~~0.9970~~
R^2	0.3325	0.0220~~0.0025~~	0.5512	0.310	0.1301	5.7E-03	0.068~~.0582~~
Shapiro-Wilk Pass	No	No	Yes	Yes	Yes	No	No

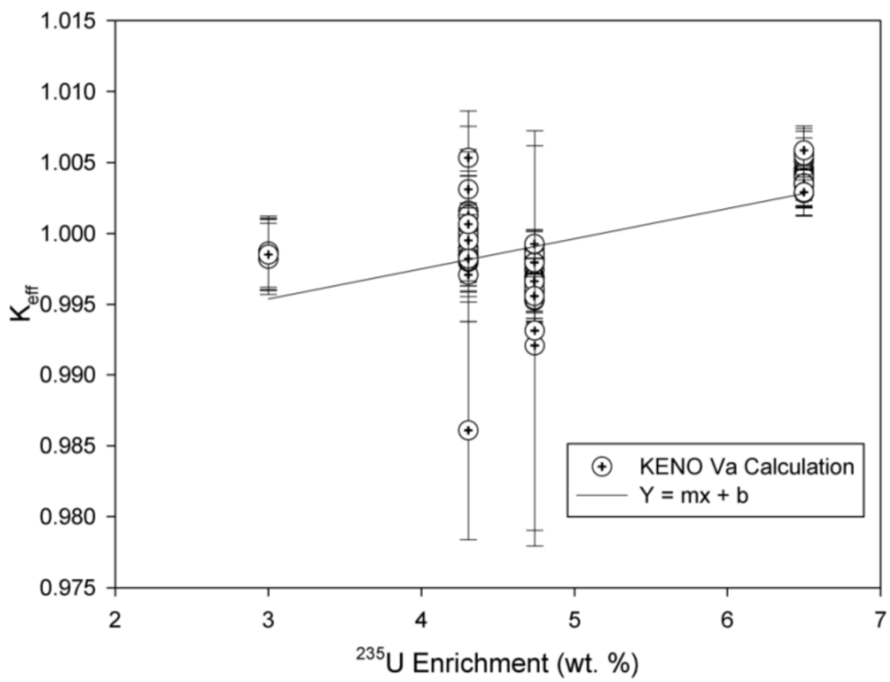

Figure 3—205 Regression analysis of U-235 enrichment

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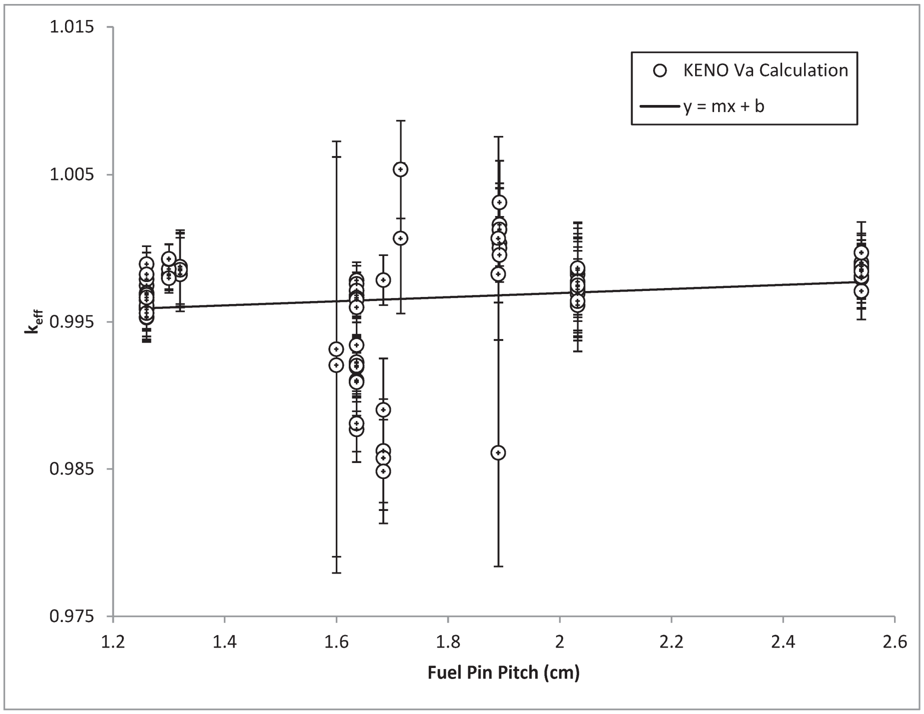

Figure 3—206 Regression analysis of fuel rod pitch

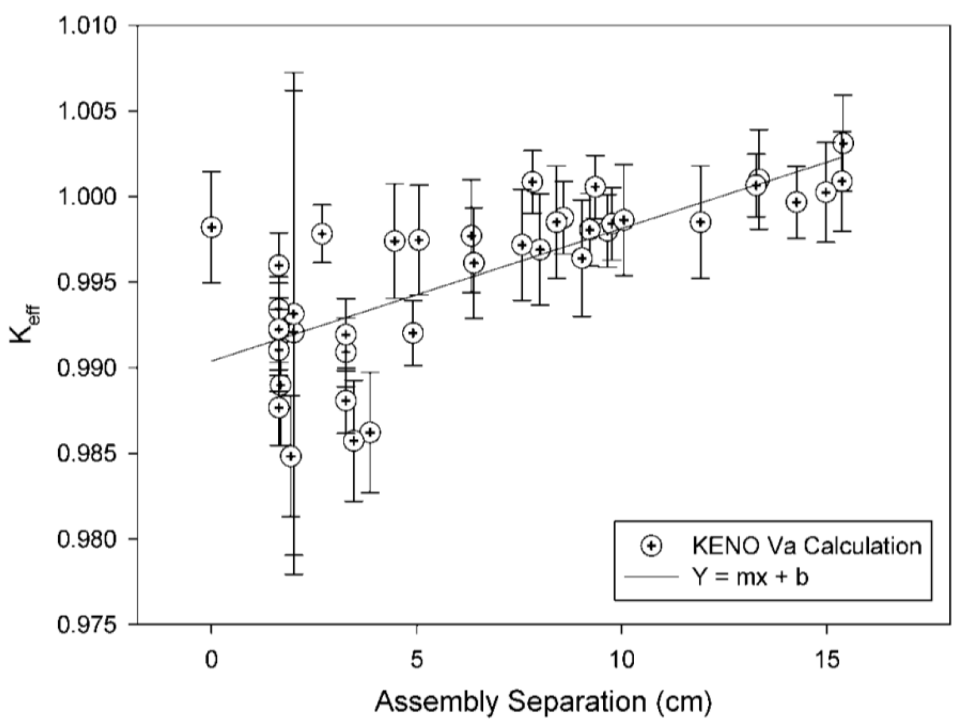

Figure 3—207 Regression analysis of fuel assembly separation

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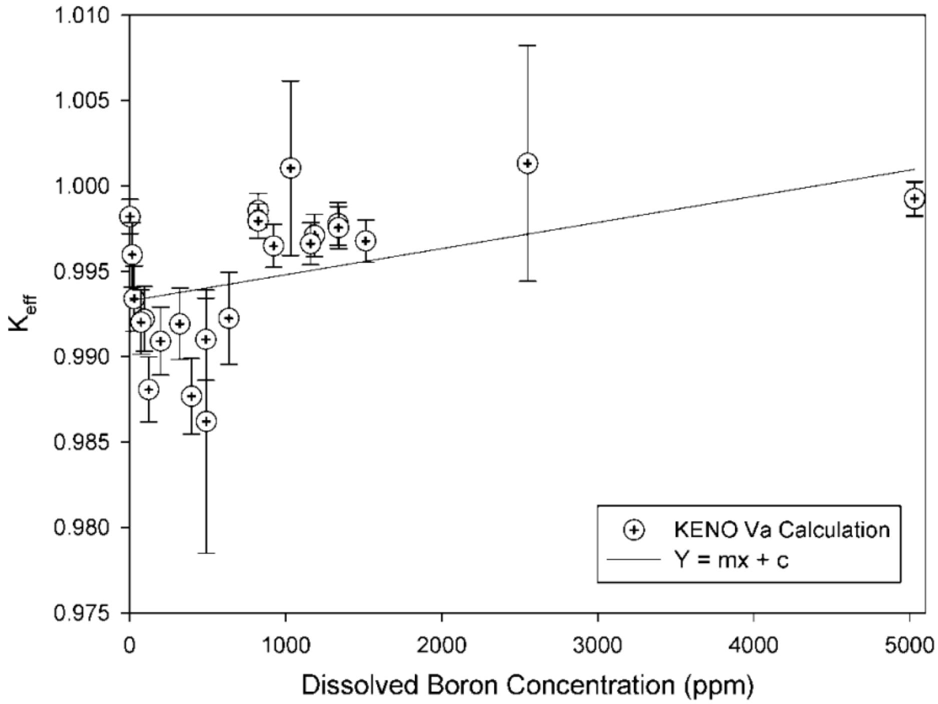

Figure 3—208 Regression analysis of dissolved boron concentration

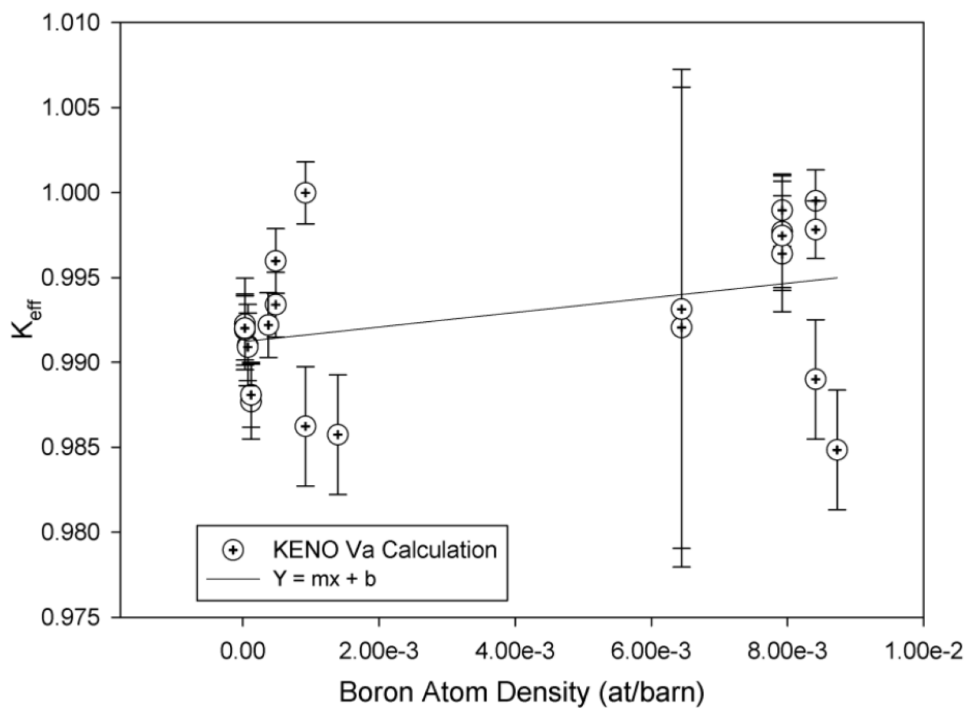

Figure 3—209 Regression analysis of separator plate areal B-10 density

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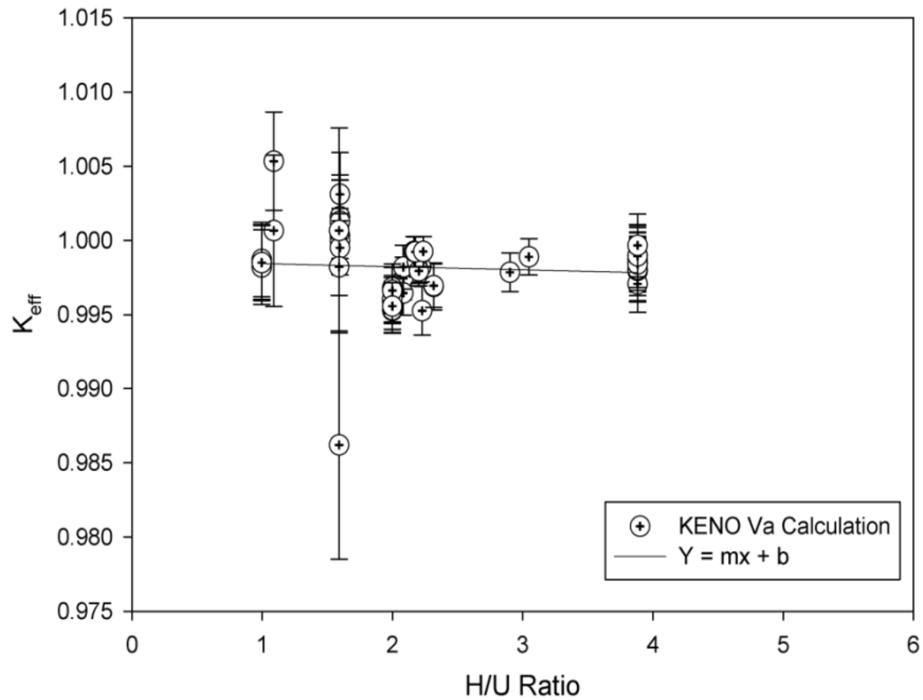

Figure 3—210 Regression analysis of moderator to fuel area ratio

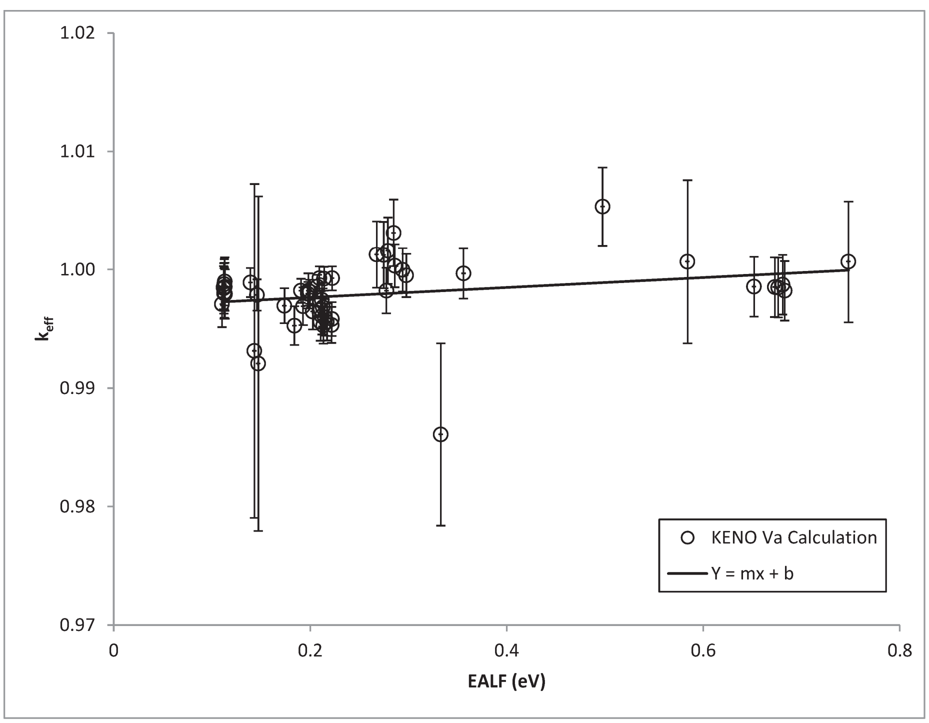

Figure 3—211 Regression analysis of neutron spectrum

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3.3.7.7 Benchmark Summary

A total of 69 benchmark critical experiments are used to establish the bias and uncertainty of the bias for use in the criticality safety analysis of the fuel storage racks. Additional experiments are used to facilitate the determination if any valid biases are present in the benchmark data.

Trending analyses are performed on seven parameters using linear regression. In general, the analysis shows very small slopes (indicating little to no change in the k_{norm} with changes to the independent variable), poor correlation coefficients, and significant fitting parameter uncertainty. As a result, there is no basis to justify a modification to the calculated bias as a result of any trend.

The bias is -0.00064 and the bias uncertainty is 0.00676.

3.3.7.8 Implementation and Use

The results presented in Section 3.3.7.7 are applicable to the fuel storage racks over the area of applicability given in Table 3-89.

The area of applicability for enrichment is 3 to 6.5 wt% U-235, which covers the limiting enrichment of 5 wt% U-235 for the NuScale analysis. FAs lower than 3 wt% U-235 are allowed without penalty on the bias or bias uncertainty because the lower-enrichment fuel is not limiting.

The area of applicability for neutron spectrum measured by EALF is 0.11~~0.0198~~ to 0.748 eV. No additional penalty is required on the bias or the bias uncertainty because the calculated k-effective becomes less limiting as the moderator density decreases.

Table 3-89 Area of applicability for bias and bias uncertainty

Parameter	Range	Estimated NuScale Value
U-235 enrichment	3 – 6.5 wt%	5 wt%
Fuel rod pitch	1.26 – 2.54 cm	1.26 cm
FA separation	0 – 15.393 cm	7 cm
Soluble boron concentration	0 – 5030 ppm	0 – 2000 ppm
Boron areal density in separator plates	~~3.5~~3.0E-5 - 8.7E-3 at/barn	~~4E-3~~2.8E-3 at/barn
Moderator to fuel ratio	0.996 – 3.882~~5.067~~	1.99
Neutron spectrum (EALF)	~~0.0198~~0.11 – 0.748 eV	0.3 eV

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-10

SRP Section 9.1.1 provides guidance for complying with 10 CFR Part 50, Appendix A, GDC 62 and 10 CFR 50.68. 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.

Page 2 of TR-0816-49833-P states, "[t]he fuel storage racks can safely store at least 1404 FAs [fuel assemblies] vertically in the SFP [spent fuel pool]." This statement is inconsistent with NuScale Generic Technical Specification (TS) 4.3.3, "Capacity," which states, "[t]he spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1404 fuel assemblies" (emphasis added). Thus, TS 4.3.3 specifies, and TR-0816- 49833-P should reflect, that the fuel storage racks can safely store at most 1,404 fuel assemblies in the SFP.

In addition, Page 2 of TR-0816-49833-P states that a maximum of 1,404 fuel assemblies is used in the fuel storage rack analyses. However, this is not consistent with the statement on Page 290 of TR-0816-49833-P that the fuel storage rack criticality analysis assumes all rack locations, even those that are inaccessible, are used for the whole- pool model. The staff recognizes the conservatism of assuming all rack locations being used for the whole-pool criticality model but notes that the information on Page 2 of TR-0816-49833-P should be consistent with what was actually done for the analyses.

For consistency of information in TR-0816-49833-P, please update the document as appropriate to clarify the points above. Accurate and consistent information regarding the fuel storage rack design and model is necessary for the staff to reach a reasonable assurance finding on GDC 62 and 10 CFR 50.68.

NuScale Response:

Section 1.0 of TR-0816-49833 has been revised as shown in the attached to clarify that at most 1,404 fuel assemblies can be placed in the accessible storage locations in the fuel assembly storage racks. In addition, the revision clarifies that the analyses in the Technical Report also consider more or fewer stored fuel assemblies as described in some of the analyses.

Impact on DCA:

Technical Report TR-0816-49833, Fuel Storage Rack Analysis, Section 1.0, has been revised as described in the response above and as shown in the markups provided with this response.

1.0 Introduction

The fuel storage racks are treated as safety-related and designed as Seismic Category 1 in accordance with requirements of 10 CFR Part 50. This report is prepared in accordance with the guidance provided by the NuScale Design Specific Review Standard (DSRS).

The fuel storage racks are designed as modular rack systems for storing, loading, and unloading fuel assemblies (FAs) in the spent fuel pool (SFP). The design incorporates a flux trap and fixed neutron absorbers. The racks are able to store both irradiated and un-irradiated fuel in the water-filled SFP. The design is a general placement design developed for the long term storage of irradiated fuel and new (unirradiated) fuel. The analyses performed do not impose any restriction for FA placement in the SFP. The fuel storage rack is an 11x11 configuration, allowing up to 121 FAs to be safely stored per rack.

The fuel storage racks can safely store at ~~least~~most 1404 FAs vertically in the SFP, factoring in the maximum reach of the fuel handling machine path. Because of the reach of the fuel handling machine (red line shown in Figure 1—2), not all fuel cells can safely be reached. Therefore, a maximum of 1404 FAs ~~is used in the analyses~~can be placed in the accessible storage locations but more or fewer stored FAs are also considered in some of the analyses as described below.

1.1 Abbreviations

Table 1-1 Abbreviations

Term	Definition
ALE	Arbitrary-Lagrangian-Eulerian
ASME	American Society of Mechanical Engineers
BOL	beginning of life
CEUS	Central and Eastern United States
CFD	computational fluid dynamics
CRA	control rod assembly
DSRS	Design Specific Review Standard
EOL	end of life
FA	Fuel Assembly
FEA	finite element analysis
HF	high frequency, refers to high-frequency target-response spectra, i.e., HF1 and HF2
HX	heat exchanger

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8760
Date of RAI Issue: 04/25/2017

NRC Question No.: 09.01.01-11

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 3.3.6.6, "Fuel Storage Racks Seismic Event," states that there is a transient deflection of no more than 0.010 inch per storage tube. However, Table 3-22, "Maximum fuel assembly gap reduction," in Section 3.1.4.10.6 shows that the maximum fuel assembly gap reduction is {{ }}[2(a)(c)], more than twice what is stated in Section 3.3.6.6.
b. Table 3-77, "Parameter range for critical experiment selection," makes it appear as though the criticality analyses used a moderator density of 1.0 g/cm^3, while they actually used a density of 0.9982 g/cm^3 according to Page 286.
c. Section 3.3.7.1 appears to contain several incorrect references:
    i. In Section 3.3.7.1.1, NUREG/CR-6361 is referenced, while the equations actually come from NUREG/CR-6698
    ii. In Section 3.3.7.1.2, ANSI N15.15-1974 is listed as Reference 21 but should be Reference 22 according to the References list
    iii. The equations listed in Section 3.3.7.1.2 appear to come from ANSI N15.15-1974, not NUREG/CR-6361
d. Section 3.3.7.2 appears to contain some incorrect references and information:
    i. Experiment LEU-COMP-THERM-001 only contains pitches of 2.032 cm (not 1.684 cm)
    ii. Experiment LEU-COMP-THERM-002 only contains pitches of 2.54 cm (not 1.892 cm)
    iii. Experiment LEU-COMP-THERM-010 is from Ref. 23, not 22
    iv. Experiment PAT80 is from Ref. 21, not 23

e. In Section 3.3.7.6, "Bias and Bias Uncertainty," the terms appear to be reversed in the equation for the bias
f. In Table 3-90, "Summary of criticality analysis results," the following items appear to be inconsistent with other information in the report:
   i. $k_{95/95}$ when flooded with unborated water at full moderator density is listed as 0.94440, not 0.94441, in Table 3-69, "System sensitivity to nominal temperature"
   ii. In the last row of the table, the cited seismic deformation of 0.010 is inconsistent with the {{ }}$^{2(a)(c)}$ listed in Section 3.1.4.10.6.

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

---

NuScale Response:

The Technical Report on Fuel Storage Rack Analysis, TR-0816-49833, Revision 0, has been revised to be accurate and consistent as described in the responses below:

Part a:

The value for the deflection is correct in Table 3-22, "Maximum fuel assembly gap reduction." Section 3.3.6.6 and Table 3-90 have been revised as shown in the attached markups for consistency with Table 3-22.

Part b:

Table 3-77, "Parameter range for critical experiment selection," shows the correct critical parameter range of moderator densities used in the selection of critical experiments for the benchmark cases, i.e., ±10 wt% of 1.0 g/cm^3. However, that range was not used in the NuScale analysis but just for selection of the critical experiments that benchmarked the validity of the computational methods for fuel storage in the racks. As shown in the response to RAI Question 09.01.01-4, additional clarification was added in TR-0816-49833 to explain that a bias term was included in the calculation of $k_{95/95}$to adjust the system operating temperature to 40 degrees F and a density of 1.0 g/cm^3.

Part c:

The changes to references identified by the RAI Question are correct. Section 3.3.7.1, "Methodology," has been revised as follows and shown in the attached markups:

i. In Section 3.3.7.1.1, "Bias and Bias Uncertainty," references to NUREG/CR-6361 have been revised to the correct reference, NUREG/CR-6698. Also, the number of an equation from NUREG/CR-6698 was revised to the correct number.
ii. In Section 3.3.7.1.2, "Normality," the reference number for ANSI N15.15-1974 was

revised to Reference 22 to be consistent with Section 5.2, "References Cited." Also, the reference to NUREG/CR-6361 (Reference 21) was corrected to NUREG/CR-6698 (Reference 20).

iii. In Section 3.3.7.1.2, "Normality," citations to NUREG/CR-6361 have been revised to the correct reference, ANSI N15.15-1974.

Part d:

The changes for references and information identified by the RAI Question are correct. Section 3.3.7.2, "Selection of Experiments," has been revised as follows and shown in the attached markups:

i. In Table 3-78, "All critical experiments used for bias determination or trending or both," the information for Experiment LEU-COMP-THERM-001 has been revised to identify only a pitch of 2.032 cm.
ii. In Table 3-78, Experiment LEU-COMP-THERM-002 has been revised to identify only a pitch of 2.54 cm.
iii. In Table 3-78, the reference number for Experiment LEU-COMP-THERM-010 has been revised to Reference 23.
iv. In Table 3-78, the reference number for Experiment PAT80 has been revised to Reference 21.

Part e:

The change in the order of the terms in the equation for the bias as identified by the RAI Question is correct. Section 3.3.7.6, "Bias and Bias Uncertainty," has been revised and placed in the correct order as shown in Section 3.3.7.1.1, "Bias and Bias Uncertainty," and in the attached markups.

Part f:

Table 3-90, "Summary of criticality analysis results," has been revised as follows:

i. The data presented in Table 3-69 is based on a single rack calculation. Table 3-90 is based on full pool analyses. Table 3-69 was updated based on RAI Question 09.01.01-4. Table 3-90 was updated based on the additional bias to address RAI Question 09.01.01-2. See the markups for the response the RAI Question 09.01.01-2.
ii. The value for deformation was corrected in response to RAI Question 09.01.01-11, Part a, as shown in the attached markups.

Impact on DCA:

Technical Report TR-0816-49833, Rev. 0, Fuel Storage Rack Analysis, Sections 3.3.6.6, 3.3.7.1.1, 3.3.7.1.2, 3.3.7.2, and 3.3.7.6; and Tables 3-78 and 3-90 have been revised as described in the response above and as shown in the markups provided with this response.

Table 3-75 Spent fuel pool dropped fuel assembly analysis

Scenario	k_{eff}	σ	$k_{95/95}$
Full pool, no dropped fuel	0.86697~~0.86684~~	0.00009~~0.00031~~	0.90193~~0.89949~~
1 dropped fuel assembly in fuel elevator area	0.86714~~0.86668~~	0.00014~~0.00026~~	0.90210~~0.89933~~

3.3.6.5 Misloaded Fuel Assembly

The fuel storage analysis conservatively assumes the racks are completely loaded with FAs at the maximum reactivity. There are no restrictions on loading patterns, therefore, there is no possibility of misloading an assembly in the fuel storage racks.

3.3.6.6 Fuel Storage Racks Seismic Event

The mechanical analysis of the fuel storage racks for a seismic event demonstrates that the racks do not undergo permanent deformation. However, there is a transient deflection of no more than ~~0.010 inch~~{{ }}$^{2(a),(c)}$ per storage tube (see Section 3.1.4.10.6). The seismic event is analyzed with the single fuel storage rack model and periodic radial boundary conditions. The spacing between storage tubes is reduced in a series of cases to determine the maximum deformation that maintains $k_{95/95}$ less than the limit of 0.95. The results are shown in Table 3-76 and Figure 3—204. The values of $k_{95/95}$ shown in this table are calculated using Equation 1 in Section 3.3.1, using the appropriate values shown in Table 3-74. {{

}}$^{2(a),(c)}$

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3.3.7.1 Methodology

3.3.7.1.1 Bias and Bias Uncertainty

When comparing the experimentally measured k-effective (k_{exp}) to the calculated k-effective (k_{calc}), the values are normalized as shown in NUREG/CR-6698~~6361~~, Equation 9 (Reference 20):

$$k_{\text{norm}} = {k_{\text{calc}}}\Big/{k_{\text{exp}}}$$

The measured k-effective is adjusted for some experiments where the published benchmark model has a known omission, which causes a small but non-negligible modeling bias.

In addition, the errors are combined statistically as shown in NUREG/CR-6698~~6361~~, Equation 3~~9~~:

$$\sigma_t = \sqrt{\sigma_{\text{calc}}^2 + \sigma_{\text{exp}}^2}$$

The weighted mean value of k-effective is calculated with the following set of equations as shown in NUREG/CR-6698~~6361~~, Equations 4 – 7:

Variance about the mean:

$$s^2 = \frac{\left(\frac{1}{n-1}\right)\sum \frac{1}{\sigma_i^2}\left(k_{\text{norm}_i} - \overline{k_{\text{norm}}}\right)^2}{\frac{1}{n}\sum \frac{1}{\sigma_i^2}}$$

Average total uncertainty:

$$\overline{\sigma^2} = \frac{n}{\sum \frac{1}{\sigma_i^2}}$$

Weighted mean k_{norm}:

$$\overline{k_{\text{norm}}} = \frac{\sum \frac{1}{\sigma_i^2} k_{\text{norm}_i}}{\sum \frac{1}{\sigma_i^2}}$$

Square root of the pooled variance:

$$S_p = \sqrt{s^2 + \bar{\sigma}^2}$$

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Finally, the bias is determined as:

$$\text{Bias} = \overline{k_{\text{norm}}} - 1$$

if k_{norm} is less than one; otherwise Bias = 0.

And the uncertainty on the bias is

$$C * S_p$$

Where C is the 95/95 tolerance factor dependent on the sample size.

3.3.7.1.2 Normality

The use of the pooled variance and the single-sided tolerance limit is predicated on the requirement for the data to be representative of a normal distribution. In the details given by NUREG/CR-6698~~6361~~ (Reference 20~~21~~), a Shapiro-Wilk test is used. While appropriate for that given example, the data set in this document has a much larger population. The Shapiro-Wilk test, which is optimum for samples of 50 or less, is not appropriate for the data gathered for this application. A more robust test, D'Agostino (from ANSI N15.15-1974 (Reference 22~~21~~)), is used to verify that the sampled experiments are representative of a normally distributed data set. Evaluating the test statistic ANSI N15.15-1974~~NUREG/CR-6361~~, Section 7.2.4:

$$D' = \frac{T}{S}$$

Where ANSI N15.15-1974~~NUREG/CR-6361~~, Section 7.2.2:

$$\mathrm{T} = \sum_{\mathrm{i}=1}^{\mathrm{n}} \left[\mathrm{i} - \frac{(\mathrm{n}+1)}{2}\right] \mathrm{x_i}$$

And ANSI N15.15-1974~~NUREG/CR-6361~~, Section 4.2.2:

$$S^2 = \sum x_i^2 - \frac{\sum (x_i)^2}{n}$$

Critical values for the 95th percentile would be P(0.025) and P(0.975) for the two-tailed test.

3.3.7.2 Selection of Experiments

A large set of experiments is selected for either the determination of the bias and bias uncertainty, or for the determination of trends, or for both uses. The complete list of experiments and the description of the experiments is provided in Table 3-78.

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Table 3-78 All critical experiments used for bias determination or trending or both

Experiment Name	Reference	Description
LEU-COMP-THERM-001	Ref. 23	A series of critical approach experiments with clusters of aluminum-clad UO2 fuel (2.35 wt%) rods in a large water-filled tank is performed over the course of several years at the Critical Mass Laboratory at the Pacific Northwest National Laboratories (PNNL). Experiments included rectangular, square-pitched lattice clusters, with a pitch~~pitches~~ of 2.032 cm~~ or 1.684 cm~~.
LEU-COMP-THERM-002	Ref. 23	A series of critical approach experiments with clusters of Al-clad UO2 (4.306 wt%), in a water-filled tank, performed at PNNL. Fuel is arranged in a square-lattice cluster with a pitch of~~pitches at~~ 2.54 cm~~ or 1.892 cm~~.
LEU-COMP-THERM-008	Ref. 23	Low-enriched fuel rods (2.459 wt%), clad in aluminum arranged in a square pitch of 1.636 cm. Soluble boron, ranging from 921 ppm to 1511 ppm is present in the moderator. Performed by B&W Lynchburg Research Center.
LEU-COMP-THERM-009	Ref. 23	A series of critical approach experiments with clusters of Al-clad UO2 (4.306 wt%), in a water-filled tank, performed at PNNL. Fuel is arranged in a square-lattice cluster with a pitch of 2.54 cm. Absorber plates are used to show the effect of steel, Boral, Al and others.
LEU-COMP-THERM-010	Ref. 23~~22~~	A series of critical approach experiments with clusters of Al-clad UO2 (4.306 wt%), in a water-filled tank, performed at PNNL. Fuel is arranged in a square-lattice cluster with pitches at 2.54 cm or 1.892 cm. Reflecting walls of steel are investigated in this series of experiments.
LEU-COMP-THERM-012	Ref. 23	A series of critical-approach experiments with clusters of 36-inch-long aluminum-clad UO2 fuel rods (2.35 wt%) in a large water-filled tank is performed over the course of several years at PNNL. Experiments included square-pitched lattice clusters with pitches of 2.032 cm. Effects of absorber plates are also investigated.

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Experiment Name	Reference	Description
LEU-COMP-THERM-070	Ref. 23	This is a series of experiments investigating the physics of the water-water energy reactor (VVER). The UO2 fuel (6.5 wt%) is arranged in a hexagonal lattice at a pitch of 11 mm.
LEU-COMP-THERM-075	Ref. 23	This is a series of experiments investigating the physics of the VVER. The UO2 fuel (6.5 wt%) is arranged in a hexagonal lattice at a pitch of 11 mm. Variations made include water holes and absorber rods.
PAT80	Ref. 21~~23~~	4.74 wt% UO2 fuel rods in square lattices of 1.6 cm pitch, with addition of boron separator plates

3.3.7.2.1 Selection of Experiments for Bias and Uncertainty

The subset of experiments that is used to determine the bias and the uncertainty on the bias are shown in Table 3-79.

Table 3-79 Benchmark experiments selected for bias and bias uncertainty

Experiment Name	Reference	Cases Selected	# of Cases
LEU-COMP-THERM-002	Ref. 23	1 – 5	5
LEU-COMP-THERM-009	Ref. 23	1 - 4, 9	5
LEU-COMP-THERM-010	Ref. 23	5, 16 - 19	5
LEU-COMP-THERM-013	Ref. 23	1, 2, 3	3
LEU-COMP-THERM-014	Ref. 23	1, 2, 5, 6, 7	5
LEU-COMP-THERM-039	Ref. 23	1 – 17	17
LEU-COMP-THERM-048	Ref. 23	1 – 5	5
LEU-COMP-THERM-050	Ref. 23	1 – 7	7
LEU-COMP-THERM-070	Ref. 23	1 – 12	12
LEU-COMP-THERM-075	Ref. 23	4, 5, 6	3
PAT80	Ref. 21	SS1, SS2	2
Total Cases			**69**

3.3.7.2.2 Selection of Experiments for Enrichment Trend

The experiments shown in Table 3-80 are chosen to sample the calculation of k-effective with U-235 enrichment that is representative of the fuel storage racks. The range of U-235 enrichments was from 3.0 wt% to 6.5 wt%. While this is slightly outside the desired area of applicability (3.5-6.5 wt%), it does not adversely bias the results with its inclusion, rather it lends statistical significance to the sample.

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#	Name	knorm	σt
53	LEU-COMP-THERM-070-001	1.00511	0.0016
54	LEU-COMP-THERM-070-002	1.00455	0.0016
55	LEU-COMP-THERM-070-003	1.00437	0.0016
56	LEU-COMP-THERM-070-004	1.00430	0.0016
57	LEU-COMP-THERM-070-005	1.00413	0.0016
58	LEU-COMP-THERM-070-006	1.00359	0.0016
59	LEU-COMP-THERM-070-007	1.00346	0.0016
60	LEU-COMP-THERM-070-008	1.00395	0.0016
61	LEU-COMP-THERM-070-009	1.00341	0.0016
62	LEU-COMP-THERM-070-010	1.00353	0.0016
63	LEU-COMP-THERM-070-011	1.00284	0.0016
64	LEU-COMP-THERM-070-012	1.00291	0.0016
65	LEU-COMP-THERM-075-004	1.00569	0.0017
66	LEU-COMP-THERM-075-005	1.00550	0.0017
67	LEU-COMP-THERM-075-006	1.00587	0.0017
68	PAT80SS1	0.99206	0.0141
69	PAT80SS2	0.99314	0.0141

3.3.7.6 Bias and Bias Uncertainty

Evaluating the equations from Section 3.3.7.1.1 on the data in Table 3-88 provides the following results:

$$\overline{k_{\text{norm}}} = 0.99936$$

$$\overline{\sigma^2} = 2.71 \times 10^{-6}$$

$$S_p = 3.39 \times 10^{-3}$$

The bias is then:

$$\cancel{1 -} \overline{k_{\text{norm}}} \cancel{=} - 1 = -0.00064$$

For a sample size of 69, a confidence interval of 0.95 and a 0.95 probability, the tolerance factor is 1.993 (Reference 23). The bias uncertainty is:

$$(1.993)(3.39 \times 10^{-3}) = 0.00676$$

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3.3.8 Summary of Criticality Evaluations

The results of the criticality analysis are summarized as follows:

- The normal and accident condition temperatures in the spent fuel pool range from 40 degrees F to 212 degrees F. The criticality analysis is performed at a temperature of 67 degrees F, and the $k_{95/95}$ calculated including additional bias and uncertainty for a minimum temperature of at 40 degrees F.

- The maximum $k_{95/95}$ with full density moderation in unborated water remains below the applicable limit of 1.0 for fuel stored in the storage racks without credit for burnup.

- The maximum $k_{95/95}$ with full density moderation in 800 ppm (natural boron) borated water remains below the applicable limit of 0.95 for fuel stored in the storage racks without credit for burnup.

- No accident condition is identified that would cause an inadvertent criticality.

- Considered a range of gaps and the seismic analysis confirmed that the gap is still within the allowed range and no permanent deformation occurred.

A summary of the criticality analysis results are presented in Table 3-90.

Table 3-90 Summary of criticality analysis results

Conditions	$k_{95/95}$	Limit
Flooded with unborated water at full moderator density	0.94630~~0.94441~~	1.0
Flooded with water at 800 ppm boron, full moderator density	0.90193~~0.89941~~	0.95
Flooded with water at 800 ppm boron, full moderator density, dropped FA	0.90210~~0.89933~~	0.95
Flooded with water at 800 ppm boron, rack deformation of 1.140 cm compression per cell, which is ~~(~~greater than the seismic deformation of {{ }}2(a),(c)~~0.010 in)~~	0.94157~~0.93926~~	0.95

The criticality analysis was completed without consideration for loading patterns or zoning consideration. Based on the conservative assumptions used in the criticality analysis, it is acceptable to place fuel in the racks in all cell locations.

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Enclosure 3:

Affidavit of Zackary W. Rad, AF-0617-54655

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, 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 Request for Additional Information response reveals distinguishing aspects about the component or method by which NuScale develops its fuel storage rack design.

   NuScale has performed significant research and evaluation to develop a basis for this the component or method 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 enclosed Request for Additional Information No. 04, eRAI 8760, section 09.09.01. The 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 6/26/2017.

Zackary W. Rad