Modeling & Analysis of Core Debris Recriticality During Hypothetical Severe Accidents in the Advanced Neutron Source Reactor*

by

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MODELING AND ANALYSIS OF CORE DEBRIS RECRITICALITY DURING HYPOTHETICAL SEVERE ACCIDENTS IN THE ADVANCED NEUTRON SOURCE REACTOR

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ABSTRACT

This paper discusses salient aspects of severe-accident-related reactivity modeling and analysis in the Advanced Neutron Source (ANS) reactor. The development of an analytical capability using the KENO5A-SCALE system is described including evaluation of suitable nuclear cross-section sets to account for the effects of system geometry, mixture temperature, material dispersion and other thermal-hydraulic conditions. Benchmarking and validation efforts conducted with KENO5-SCALE and other neutronic codes against critical experiment data are described. Potential deviations and biases resulting from use of the 16-group Hansen-Roach library are shown. A comprehensive test matrix of calculations to evaluate the threat of a criticality event in the ANS is described. Strong dependencies on geometry, material constituents, and thermal-hydraulic conditions are described. The introduction of designed mitigative features are described.

I. INTRODUCTION

Oak Ridge National Laboratory's (ORNL's) ANS reactor will be a new user facility\(^1,2\) for all kinds of neutron research, centered around a research reactor of unprecedented \((-10^{20} \text{ m}^{-2}\cdot\text{s}^{-1})\) neutron flux available to the beam tubes. A defense-in-depth philosophy has been adopted. In response to this commitment, ANS Project management has initiated severe accident analysis and related technology development early-on in the design phase. This was done to aid in designing a sufficiently robust containment for retention and controlled release of radionuclides in the event of an accident. It also provides a means for satisfying on- and off-site regulatory requirements and accident-related dose exposures and for containment response and source-term best-estimate analyses for the Levels-2 and -3 Probabilistic Risk Analyses (PRAs) that will be produced. Moreover, it will provide the best possible understanding of the ANS under severe accident conditions and consequently provide insights for development of strategies and design philosophies for accident mitigation, management, and emergency preparedness efforts.

This paper describes salient aspects of the work done to date on addressing a potentially important severe accident issue dealing with reactivity during hypothetical severe accidents.

A. ANS System Design

The ANS is currently in the conceptual design stage. As such, design features of the containment and reactor system are evolving based on insights from ongoing studies. Table 1 summarizes the current principal design features of the ANS from a severe accident perspective compared to ORNL's High Flux Isotope Reactor\(^3\) (HFIR) and a commercial light-water reactor (LWR). As seen in Table 1, high-power-density research reactors can give rise to significantly different severe accident issues. Specifically, the ANS reactor will use about 15 kg of highly enriched (-93 m/o 235U) uranium silicide fuel in an aluminum matrix with a plate-type geometry and a total core mass of 100 kg. About 13 g of B\(^{10}\) burnable poison is provided in the end caps of fuel plates to reduce excess reactivity at the beginning-of-cycle (BOC) and to help shape the power distribution. Heavy water (D\(_2\)O) is used as moderator and coolant. The power density of the ANS will be about 50 to 100 times higher than that of a large LWR. A schematic representation of the reactor and cooling circuit is given in Fig. 1. The reactor core is enclosed within a core pressure boundary tube and enveloped in a reflector tank. Four inlet pipes deliver D\(_2\)O coolant upward into the core at a high velocity \((-27 \text{ m/s})\), and D\(_2\)O then enters a large stainless steel pipe before branching into several pipes leading to heat exchangers. Much of the coolant system piping is submerged in light-water pools.
B. Importance of Recriticality Issue for ANS

Recriticality during severe accidents could lead to damaging steam explosion loads. Additional fission product generation and high-energy bursts of radiation are also undesirable byproducts. The scope of recriticality in ANS under hypothetical severe accidents was motivated by the need to gage the potential for such an occurrence and by the need to consider designed mitigative features early in the design process.

II. MODELING AND PROBLEM FORMULATION FOR ANS RECIRCRITICALITY ANALYSES

During hypothetical severe accidents in the ANS, fuel plate melting may occur either with or without a flowing medium adjacent to the melting plates. Hypothetical accident conditions such as core inlet flow blockages or large pipe loss-of-coolant accidents (LOCAs) may provide such conditions. Under such circumstances, and assuming that a steam explosion does not occur, the core mass may slump and agglomerate downward into the primary coolant system piping regions. Again, experiments with melting aluminum tubes in the presence of flowing media have shown that depending on the destabilizing surface forces caused by flowing media, debris dispersal and entrainment in the flowing medium may occur. Debris dispersal also may occur in the presence of steam explosions. Such dispersion mechanisms can cause fragmented core debris to be swept into the coolant outlet piping. Hence, for ANS severe accident analysis, lumped and dispersed configurations need to be analyzed for gaging recriticality potential.

The process of modeling and analysis for recriticality under severe accident conditions for ANS involved several steps. First, a modeling capability was developed to account for ANS debris in various configurations and surrounded with various geometries and materials. The modeling framework was benchmarked and validated against known critical experiments. Next, the potential for recriticality in the ANS under severe accident conditions was analyzed. Because of the absence of a mechanistic core-melt-progression capability, the various geometries and thermal-hydraulic conditions were postulated and analyzed parametrically. Again, the time-dependent behavior of the system following a recriticality event would require development of a transient modeling capability, which was considered beyond the scope of this simplified study. Hence, a wide range of parametric studies was conducted to gage the behavior of the system under various conditions of temperature and void fraction. Finally, evaluations were made for incorporation of designed mitigative features for prevention of recriticality. Salient aspects of these various steps are described subsequently.

A. Modeling Framework Development, Benchmarking, and Validation

Because of its versatility, the well known KENO5A-SCALE5 neutronic code system was the modeling framework of choice for evaluating the recriticality potential of ANS core debris. To gage applicability of this system to ANS debris lumped or dispersed geometries, a series of experiments was researched. These experiments considered lumped and dispersed fuel configurations in the presence of light and heavy water. In addition to using KENO5A for evaluating $k_{\text{eff}}$, transport-theory–based (viz., XSDRNPM, TORT6) calculations also were conducted to provide a basis for bias determination and for evaluating the appropriateness of using the 39- and 99-group ANS-L-V cross-section libraries. Several details of the comparison are omitted because of space considerations. An abstract of the benchmarking and validation exercises are presented.

Results of KENO5A and XSDRNPM calculations for $k_{\text{eff}}$ against the well-known GODIVA8 bare enriched U-metal sphere experiment are given in Table 2. Note that criticality is evaluated within one standard deviation of the experiment. This forms an important benchmark for the cross-section libraries used in analysis of ANS reactor lumped-core cases. Thereafter, ancillary calculations were performed with XSDRNPM and KENO5A for three H2O-reflected spheres and five D2O-reflected spheres. The results of XSDRNPM and KENO5A agreed within one standard deviation of individual calculations, representing excellent comparisons.

Five ORNL critical spheres9 consisting of enriched uranyl nitrate in water in one of two spheres of radius 345.98 or 610.108 mm (the first four spheres having the smaller radius) were analyzed. The first and fifth spheres contain the critical concentrations for unpoisoned solutions within the two spheres. In the other three, criticality is maintained by counterbalancing increases in the uranium concentration with increases in the natural boron concentration of the solution. Selected results are presented in Table 2. The results of computations are within 0.25% uncertainty of the experiments, and again, excellent agreement is seen between predictions and experiment, and between XSDRNPM and KENO5A calculations. Also, note that use of 39- and 99-group cross-sections gives almost identical results. Separately, additional comparisons also were made against six D2O-reflected spheres10 and five bare cylinders filled with uranyl fluoride in heavy water using XSDRNPM and DORT. Excellent agreement was obtained between predictions and experiments. Separately, five supplemental ORNL-reflected and bare critical spheres consisting of enriched uranyl fluoride in water were analyzed. These experiments supplemented the earlier uranyl nitrate sphere comparisons in that a wider range of
$^{235}\text{U}$ ratios was introduced, and reflected spheres were included. Predictions were made with both the ANS-L-V 39- and 99-group cross-section libraries in the SCALE system and with XSDRNPM to perform k-calculations. As noted in Table 2, calculated results were found to be in excellent agreement with experimental data and confirm applicability of the ANS-L-V cross-section libraries in the SCALE system. Together, the above-mentioned comparisons provide reasonable confidence in the utility of the KENO5A-SCALE system for evaluation of $k_{\text{eff}}$ values for ANS core debris in either lumped or dispersed configurations.

The KENO5A-SCALE system was ported to work on an IBM/RISC-6000 workstation platform. Twenty-five benchmark calculations were executed using the integrated system. Excellent agreement was obtained between KENO5A-SCALE on the ORNL mainframe and on the workstation computers.

B. Establishing a Bias for the Calculated $k_{\text{eff}}$

Because, for a critical system, $k_{\text{eff}} = 1.0$, deviations of the calculated values from unity indicate some bias in the calculational methods and/or data. To ensure subcriticality of the ANS core debris, the calculated $k_{\text{eff}}$ values should be below established limits. The established limits are set at $k_{\text{avg}} - 3\sigma - 0.02$ (where 0.02 has been subtracted for extra shutdown margin). From comparison of the SCALE system results for $k_{\text{eff}}$ for both lumped and dispersed configurations one arrives at bias $k_{\text{eff}}$'s of 0.965 for lumped and dispersed geometries. Thus, any configuration with a calculated $k_{\text{eff}}$ greater than 0.965 would be considered critical.

III. ANS DEBRIS RECRIPTICALITY MODELING AND ANALYSIS

The ANS system is being developed. Previous core designs consisted of ~26 kg of $^{235}$U, which now has been reduced to 15 kg. Calculations for $k_{\text{eff}}$ with the larger fuel loading also have been conducted and are reported. To evaluate the threat of a recriticality event under different conditions, several different configurations and thermal-hydraulic conditions needed to be analyzed. The development of a suitable test matrix and analysis results are described subsequently.

A. Test Matrix Development and Modeling

The test matrix of calculations is shown in Tables 3 and 4. As noted therein, calculations for $k_{\text{eff}}$ were conducted with lumped and dispersed core debris materials in a stainless steel pipe filled with D$_2$O and reflected on the outside by H$_2$O. The lumped configurations were analyzed first and were found to be relatively unimportant. Because of the importance of dispersed configurations, most parametric calculations were conducted in dispersed geometries. As mentioned previously, lumped configurations are studied in a smaller diameter pipe [approximating the core inlet piping of 487-mm (19-in.) inner diameter and 10-mm thickness]. Dispersed configurations are studied for debris dispersal in the 610-mm (24-in.) schedule 20 outlet piping. Where not specifically indicated, a room temperature assumption was used for the calculations, for conservatism. Again, unless otherwise indicated, the core debris is assumed to be contained in a pipe volume extending over 3 m. Dispersed configurations are assumed to be nominally distributed over a length of 1 m, unless otherwise stated in Table 4.

A typical geometry for dispersed debris recriticality calculations is shown in Fig. 2. As seen therein, the modeled regions are divided into four material zones. Zone 1 comprises the fueled zone. A reflecting boundary (Zone 2) extends to a distance of 1 m from both ends of the mixing or fuel zone. The 1-m length was calculated essentially to provide infinite reflection of neutrons. The stainless steel piping constitutes Zone 3. Finally, the H$_2$O outside the primary coolant piping is represented as Zone 4.

As mentioned previously, lumped fuel calculations (even with 26 kg of $^{235}$U) gave rise to significantly subcritical values of $k_{\text{eff}}$. This was not true for dispersed geometries. A detailed test matrix thus was developed for parametrically evaluating the effects of changes in important variables. This is given in Table 4. The base case was developed where coolant temperature is set at 50°C to represent nominal coolant outlet temperature of the ANS core under normal operation (instead of assuming room temperature). BOC inventory of fuel is assumed for the base case, coupled with all of the 13 g of B$_{10}$ burnable poison. It was assumed that upon fuel melting-cum-dispersion, only the aluminum in the fuel meat section would accompany the fuel. Hence, about 47 kg of aluminum is associated with the base case debris recriticality calculations in Table 4. Further, it is assumed that the fuel debris would cool down to 50°C by the time the 1-m-length dispersion occurs in the outlet piping. Based on ANS technical specifications, the amount of H$_2$O contamination in the D$_2$O is specified to a mole fraction of 0.002 [i.e., assuming no influx of reactor pool H$_2$O in the reactor coolant system (RCS)].

Under certain circumstances, it is conceivable that H$_2$O fraction in the primary coolant circuit may increase (e.g., LOCA or inleakage). An increase in H$_2$O content in the primary circuit will significantly change the neutronic characteristics associated with debris recriticality. This is caused by the significantly enhanced moderation by H$_2$O compared to D$_2$O. However, this increased moderation characteristic is compensated by higher absorption. To
capture such interactions, several mole fractions of H$_2$O are included in the test matrix.

An important parameter that can significantly affect reactivity is the degree of voiding generated in the coolant. Such voiding may initiate in a previously cold system that has become critical, whereby the fuel material heats up to cause coolant boiling. For purposes of modeling, voids are assumed to be homogeneously distributed in Zone 1 (i.e., with fuel, D$_2$O, and H$_2$O). As is well known, increased voiding can provide negative reactivity feedback, which then may shut down the criticality escalation. Several KENO5-SCALE calculations are included in the test matrix. For these calculations, the fuel mixture (i.e., U, Si, B, Al) is assumed to be at the aluminum melting temperature of 660°C. Herein, a conservative assumption is made that heat from the core debris goes only towards changing phase in the coolant. However, the coolant temperature also may increase. Therefore, additional calculations are included in the test matrix with Zone 1 contents equilibrated at 72°C (i.e., perfect mixing of fuel at melting temperature with 281 kg of D$_2$O). An additional case considered the entire mixture at 100°C. This was done to represent a possible situation wherein molten core debris may have superheated above the aluminum melting temperature, and then mixed with the D$_2$O coolant to reach boiling conditions at atmospheric pressure. These calculations evaluate the effect of temperature on reactivity. The effect of temperature on fuel arises mainly from Doppler broadening. Hence, the fuel temperature coefficient of reactivity is determined mainly by resonance absorption. Because moderator density decreases with increasing temperature, the moderator coefficient of reactivity may be attributed to the change in thermal utilization. For these calculations, densities of D$_2$O and H$_2$O are suitably changed with temperature to account for the appropriate reduction in number densities of hydrogen, deuterium, and oxygen atoms. Densities of other materials are assumed to remain unchanged.

As mentioned previously, the 1-m length of the fuel debris mixture (i.e., Zone 1) was chosen arbitrarily. Clearly, a change in this length will cause the D/U ratio to change. Therefore, system criticality also can be significantly affected. Hence, parametric studies are conducted for different dispersion lengths.

Different amounts of aluminum may accompany the fuel debris in a severe accident. Hence, calculations are conducted to account for this effect.

Also, a severe accident-induced debris recriticality may occur at the end of the cycle (EOC) when ~30 to 40% of the $^{235}$U and all of the $^{210}$Bi are depleted. These cases also are studied conservatively, assuming the absence of fission product poisoning. Note that the EOC case with about 40% $^{235}$U depleted would also tend to represent a case wherein only the unirradiated outer fuel element undergoes a hypothetical severe accident-induced core debris dispersion (albeit without any burnable poison).

### B. Analysis Results

Specific KENO5 models for the various cases in Table 3 were set up and executed. The results of the $k_{\text{eff}}$ calculations are summarized in the tables and are shown graphically in Figs. 3 through 8. Unless otherwise stated, all calculations were conducted with the KENO5-SCALE system using the 39-group cross-section library.

As noted in Table 3, even if the U$_3$Si$_2$-Al mixture consisting of 24 kg of $^{235}$U were to form a lump in the inlet pipe region, the system remains significantly subcritical (i.e., $k_{\text{eff}} = 0.873$). For the same pipe geometry, a dispersed configuration also was evaluated. As can be seen, the dispersed geometry leads to a $k_{\text{eff}}$ value significantly greater than 1.0. These same calculations also were conducted using the well known Hansen-Roach library$^{12}$ [with suitable adjustments for selecting the resonance self-shielding cross-sections ($\sigma_p$), as recommended in Ref. 13]. As seen in Table 3, although the predicted $k_{\text{eff}}$ values are in the general vicinity of the ones predicted using 39-group library, use of the 16-group Hansen-Roach cross-section library (at least under these conditions) can lead to significant underprediction of $k_{\text{eff}}$. Because the ANS fuel mass is critical in the core region, it can be surmised that the significantly reduced value for $k_{\text{eff}}$ in lumped configurations results from the poisoning effect of the stainless steel piping. This aspect was confirmed via additional calculations, wherein the steel pipe was modified to be made with aluminum. Significantly higher values for $k_{\text{eff}}$ were noted (i.e., by more than 7%).

Because the lumped core-debris configuration with 26 kg of $^{235}$U remained significantly subcritical, it was decided that lumped configurations in the ANS RCS would not lead to a recriticality threat. Therefore, only dispersed configurations were studied further with the current fuel loading in the ANS core (viz., 15 kg of $^{235}$U, as shown in Table 1). The results for individual cases are tabulated in Table 4. Figures 3 through 8 show $k_{\text{eff}}$ variation with the (D+H)/$^{235}$U atom ratio in the core debris mixing zone.

Figure 3 indicates that H$_2$O contamination in the RCS can significantly increase $k_{\text{eff}}$ values. This is because of enhanced moderation. However, the effect tapers off beyond 50% H$_2$O mole fraction and then starts to decrease because of enhanced neutron absorption. These calculations demonstrate the need to keep H$_2$O out of the RCS. Note, that for nonsevere accident conditions recriticality from H$_2$O ingress is prevented by design.
Under such circumstances, control rods immediately insert to counter reactivity addition from light water entry into the RCS or into the reflector tank.

A linear decrease in $k_{eff}$ is seen in Fig. 4 with increasing void fraction, in the debris zone. With only 20% void fraction, the system $k_{eff}$ drops from 1.04 to 0.89 (viz., a 15% decrease). The variation with increased void fraction also tends to indicate that a strong mechanism exists for limiting a reactivity excursion event. A strong variation also is seen with dispersion length in Fig. 5. Reducing dispersion length causes a lumped mass-type geometry and decreases $k_{eff}$. As seen in Fig. 5, $k_{eff}$ values do not increase significantly beyond a 1-m dispersion length. Only a relatively mild variation with mixture temperature was noted. Figure 6 shows that a $k_{eff}$ decrease of about 7–8 cents/C is achieved. This result indicates that a resonance absorption caused by Doppler - broadening would provide enough negative feedback to compensate for positive reactivity insertion from increased thermal utilization by the fuel as the temperature increases. Overall, these variations demonstrate the significance and importance of properly modeling thermal hydraulic conditions during severe accidents.

Figure 7 shows that the amount of aluminum accompanying the core debris also can have a significant effect on system criticality. The variation of $k_{eff}$ with aluminum mass is almost linear. It is not as strong as seen with variation with void fraction. However, it is significant and demonstrates the importance of proper core-melt progression modeling.

Finally, Fig. 8 demonstrates the importance of $B^{10}$ in the fuel mixture. As can be seen, under EOC-type conditions when $\sim 30$ to 40% of the $^{235}U$ and all of the $B^{10}$ are depleted, the $k_{eff}$ value goes up significantly from 1.04 to $\sim 1.11$ and then starts declining. Obviously, this variation with burnup is predicted on the $B^{10}$ accompanying the fuel debris at BOC conditions in the first place.

C. Prevention and Mitigation of Debris Recriticality

Loads in ANS

An important byproduct of the results shown in Fig. 8 deals with a possible approach for mitigation of recriticality. It demonstrates that incorporation of borated pipe regions in strategic locations could play a very important role in preventing recriticality. A preliminary calculation was conducted to demonstrate this aspect, wherein a previously supercritical configuration was made significantly subcritical by borating the ANS outlet pipe. This result is currently being studied.

Overall, it is clear that debris recriticality in the ANS RCS can be effectively prevented if dispersed configurations are avoided. These evaluations demonstrate, to the extent that they represent expected conditions, that a mechanism should be found that prevents dispersion of a large enough portion of core debris during severe accidents. If fuel dispersion is inevitable, it is clearly preferable to introduce design features that only allow small portions to disperse. Clearly, the need for prevention of debris dispersion has to be balanced with the need for maintaining debris coolability (which is enhanced with dispersion). Research efforts thus are to be focused toward analytically quantifying melt progression aspects with the potential for leading to recriticality, coupled with qualification via scaled experimentation.

All in all, this is a clear case where a design fix that will prevent recriticality is far preferable to an extensive research program that may solve the problem. This is because not much is known on modeling and analysis of "transient" debris recriticality events.

IV. SUMMARY AND CONCLUSIONS

This paper has described salient aspects of benchmarking and validation of the KENO-Scale neutronic code system for evaluation of system criticality, wherein lumped and dispersed core-debris configurations may arise during hypothetical severe accidents in the ANS. Benchmarking and validation were done against data from a series of critical experiments, as also between various codes. These comparisons demonstrated the suitability of using the KENO-Scale code system in conjunction with the 39-group cross-section library. A detailed test matrix of calculations was developed for evaluating the potential of recriticality in the ANS RCS during severe accidents. The evaluations indicated that lumped configurations in the RCS would not pose a recriticality threat. However, significant potential exists for recriticality from dispersed debris configurations. Strong dependencies were noted on key thermal-hydraulic parameters such as mixture void fraction, $H_2O$ contamination, aluminum content in debris, and dispersion length. A relatively weak dependence was noted on mixture temperature. Mixture void fraction was evaluated to be the single most important parameter affecting recriticality. These calculations indicated the importance of proper core melt progression and thermal-hydraulic modeling. It was determined that prevention of recriticality in the ANS RCS may be achieved via limitation of debris dispersion, coupled with strategic positioning of borated regions in the RCS piping. Alternate choices may also be possible (e.g., thickening of pipe walls for increased parasitic absorption, or modifying pipe diameters to stay away from optimum $D/^{235}U$ ratio regions).
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Table 1. Severe accident characteristics of the ANS and other reactor systems

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Commercial LWR</th>
<th>HFIR</th>
<th>ANS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power, MW(t)</td>
<td>2660</td>
<td>85</td>
<td>300</td>
</tr>
<tr>
<td>Fuel</td>
<td>UO₂</td>
<td>U₃O₈–AI</td>
<td>U₃Si₂–AI</td>
</tr>
<tr>
<td>Enrichment, m/o</td>
<td>2–5</td>
<td>93</td>
<td>93</td>
</tr>
<tr>
<td>Fuel cladding</td>
<td>Zircaloy</td>
<td>AI</td>
<td>AI</td>
</tr>
<tr>
<td>Coolant/moderator</td>
<td>H₂O</td>
<td>H₂O</td>
<td>H₂O</td>
</tr>
<tr>
<td>Coolant outlet temperature, °C</td>
<td>318</td>
<td>69</td>
<td>85</td>
</tr>
<tr>
<td>Avg. power density, MW/I</td>
<td>&lt;0.1</td>
<td>1.7</td>
<td>4.5</td>
</tr>
<tr>
<td>Clad melting temperature, °C</td>
<td>1850</td>
<td>580</td>
<td>580</td>
</tr>
<tr>
<td>Hydrogen generation potential, kg</td>
<td>850</td>
<td>10</td>
<td>12</td>
</tr>
</tbody>
</table>

Table 2. Comparison of code predictions with critical experiment data

<table>
<thead>
<tr>
<th>Experiment</th>
<th>Measurement k&lt;sub&gt;eff&lt;/sub&gt;</th>
<th>XSDRNPM 39-group</th>
<th>Calcs. 99-group</th>
<th>KENO calcs 39-group</th>
<th>Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>GODIVA</td>
<td>1.00 +/-0.003</td>
<td>0.9990</td>
<td>0.9979</td>
<td>0.9965</td>
<td>Bare sphere (²³⁵U mass = 49.1 kg) of radius = 87.401 mm</td>
</tr>
<tr>
<td>ORNL-1</td>
<td>1.00 +/-0.0025</td>
<td>1.0025</td>
<td>1.0012</td>
<td>1.0046</td>
<td>ORNL uranyl nitrate/water solution unreflected critical spheres of different diameters</td>
</tr>
<tr>
<td>ORNL-10</td>
<td>1.00 +/0.0025</td>
<td>1.0013</td>
<td></td>
<td>1.0031</td>
<td></td>
</tr>
<tr>
<td>L7</td>
<td>1.0000</td>
<td>1.0090</td>
<td>1.0076</td>
<td></td>
<td></td>
</tr>
<tr>
<td>L8</td>
<td>1.0004</td>
<td>1.0103</td>
<td>1.0090</td>
<td></td>
<td></td>
</tr>
<tr>
<td>L9</td>
<td>1.0000</td>
<td>1.0068</td>
<td>1.0056</td>
<td></td>
<td></td>
</tr>
<tr>
<td>L10</td>
<td>1.0000</td>
<td>1.0069</td>
<td>1.0057</td>
<td></td>
<td></td>
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<tr>
<td>L11</td>
<td>0.9999</td>
<td>1.0069</td>
<td>1.0054</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Case</td>
<td>Configuration, debris constitution, etc.</td>
<td>$k_{\text{eff}}$ calculation results</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>------</td>
<td>----------------------------------------</td>
<td>---------------------------------</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>Lumped: 220-m-diam sphere of U$_3$Si$_2$–Al at bottom of D$_2$O-filled steel pipe, submerged in H$_2$O (pipe ID = 488 mm); 24 kg $^{235}$U</td>
<td>0.873</td>
<td>0.850</td>
<td>0.866</td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>Dispersed: U$_3$Si$_2$ mass uniformly suspended in a 1-m-long section of D$_2$O-filled steel, H$_2$O submerged pipe (ID = 488 mm); 15 kg of $^{235}$U</td>
<td>1.070</td>
<td>1.030</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*These calculations were done with 39-group cross-section library in the SCALE system.

bThese calculations used the 16-group Hansen Roach cross-section library.
Table 4. Test matrix of recriticality calculations for ANS with dispersed configuration

<table>
<thead>
<tr>
<th>Case No.</th>
<th>H2O mole fraction</th>
<th>Void fraction</th>
<th>Dispersion length (m)</th>
<th>Al content (kg)</th>
<th>Temperature (°C)</th>
<th>keff</th>
<th>Standard deviation</th>
<th>Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.002</td>
<td>0</td>
<td>1</td>
<td>40</td>
<td>50</td>
<td>1.0392</td>
<td>0.0034</td>
<td>Base case</td>
</tr>
<tr>
<td>2</td>
<td>0.1</td>
<td>0</td>
<td>1</td>
<td>40</td>
<td>50</td>
<td>1.1803</td>
<td>0.0039</td>
<td>Light-water</td>
</tr>
<tr>
<td>3</td>
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4This temperature is only for U3Si2, aluminum and boron.

bThese cases are the same as the base case but with 235U depletion (30% for Case 16 and 40% for Case 17) and 100% boron depletion. Note: 40% 235U depletion corresponds to EOC.
Figure 1. Schematic Representation of ANS Reactor & Containment
zone 1 = core debris-coolant mixture
zone 2 = heavy water moderator
zone 3 = pipe wall
zone 4 = light water reactor pool

Fig. 2 Schematic representation of dispersed core debris configuration in ANS outlet pipe for recriticality evaluations
H₂O mole fraction is indicated at each corresponding data point.

Fig. 3 Effect of light water contamination on k-eff

Fig. 4 Effect of void fraction on k-eff

Fig. 5 Effect of dispersion length on k-eff

Fig. 6 Effect of fuel temperature on k-eff

Fig. 7 Effect of aluminum content on k-eff

Fig. 8 Effect of fuel depletion on k-eff