HYBRID TECHNIQUE IN SCALE FOR FISSION SOURCE CONVERGENCE APPLIED TO USED NUCLEAR FUEL ANALYSIS*

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ABSTRACT

For some criticality safety applications, reliably achieving convergence in the Monte Carlo eigenvalue simulations can be challenging. Inadequate source convergence typically corresponds to an underestimation of the effective neutron multiplication factor ($k_{eff}$) – a very troublesome possibility in the field of criticality safety. To reduce the likelihood of this possibility, a new automated source convergence sequence (Sourcerer) has been developed in SCALE to deterministically compute the fission distribution and to use it as the starting source in the Monte Carlo eigenvalue calculation. In addition to decreasing the probability of underpredicting $k_{eff}$ due to inadequate source convergence, the sequence eliminates the guesswork associated with defining an appropriate, problem-dependent starting source. Furthermore, Sourcerer can increase the efficiency of the overall simulation by decreasing the number of generations (cycles) that must be skipped prior to $k_{eff}$ accumulation. To evaluate its effectiveness, Sourcerer was applied to an actual used nuclear fuel canister that had previously demonstrated source convergence difficulties due to significant variation in loaded assembly burnup values and significant neutronic decoupling between assemblies. Compared to the common use of a starting source distributed uniformly in all fissionable regions, the use of Sourcerer increased the reliability of the $k_{eff}$ calculation for all cases in which the number of skipped cycles was below ~350; for cases with higher numbers of skipped cycles the reliability was essentially equivalent. Additionally, for a fixed uncertainty objective, the use of Sourcerer increased the efficiency of the $k_{eff}$ calculation by 71% compared to a Monte Carlo calculation with a uniform starting source.

Key Words: Used nuclear fuel, canister-specific analysis, Monte Carlo eigenvalue, fission source convergence, hybrid Monte Carlo/deterministic

1 INTRODUCTION

Used nuclear fuel (UNF) is being stored at reactor sites for longer time intervals than originally foreseen, and fuel assembly discharge burnup values are increasing. The effects of the extended storage and higher burnup values (>45 GWd/MTU) on fuel rod mechanical performance as a result of cladding material property changes are not well understood. Hence, uncertainties related to meeting regulatory criteria during extended storage of UNF and subsequent transportation are increasing. The use of more detailed canister-specific evaluations

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allows determination of more realistic safety margins that may be sufficient to offset some of these uncertainties and demonstrate that an adequate safety margin is maintained. Detailed canister-specific representations with varied radial and axial fuel material compositions can be challenging for Monte Carlo (MC) eigenvalue calculations. This is especially true when the fuel assemblies stored in the canister are neutronically decoupled by effective interstitial neutron absorbers and have large variations in burnup and associated reactivity.

The MC calculations can exhibit false convergence that causes underprediction of \( k_{\text{eff}} \) due to insufficient sampling of important portions of the system [1]. Additionally, the accumulation of \( k_{\text{eff}} \) prior to achieving source convergence can cause underprediction of \( k_{\text{eff}} \) because of the contamination of the average \( k_{\text{eff}} \) with the \( k_{\text{eff}} \) from un converged cycles [1]. A better initial estimate for the fission source distribution increases the reliability of the MC eigenvalue calculations by reducing the likelihood of producing an inaccurate \( k_{\text{eff}} \) due to inadequate source convergence. Furthermore, the use of a more accurate initial source distribution can increase the efficiency of the MC eigenvalue calculation by decreasing the number of generations (cycles) that must be skipped prior to \( k_{\text{eff}} \) accumulation.

To address the source convergence issue, the new source convergence sequence (Sourcerer) has been developed to be released in version 6.2 of SCALE [2]. Sourcerer deterministically calculates an approximate, problem-dependent fission distribution to be used as the starting source in MC eigenvalue calculations. The sequence creates an input file for a deterministic eigenvalue calculation by mapping the MC geometry on a user-defined grid and then uses the deterministic solution to generate the starting source distribution for the MC eigenvalue calculation. In previous work [3], the efficiency enhancements of this approach were demonstrated for two standard test problems devised by the OECD/NEA Expert Group on Source Convergence in Criticality-Safety Analysis [1], and the reliability improvement was demonstrated for a modified version of the “\( k_{\text{eff}} \) of the world” problem that was specifically designed to demonstrate the limitations of the current MC power iteration techniques [4]. This work analyzes the effectiveness of this new sequence on the efficiency and reliability of the calculated \( k_{\text{eff}} \) for canister-specific UNF problems.

The UNF canister system used in this analysis is a NAC-Universal Multi-purpose canister System (UMS®) that is a canister-based system designed for both storage and transportation of UNF. The canister in the NAC-UMS, called the transportation storage canister (TSC), is modeled for this analysis as being loaded with 24 Combustion Engineering 14×14 assemblies. An actual canister-specific loading pattern was used in the model. The UNF assembly characteristics consisted of average initial fuel enrichments ranging from 2.0 to 2.96 wt% \(^{235}\text{U}\), average assembly discharge burnups ranging from 6.5 to 29.9 GWd/MTU, discharge dates between 1974 and 1980, and different cooling times until March 1\(^{\text{st}}\) 2004 [5]. For reactivity control, the internal basket of this TSC implements a flux trap configuration that is very effective in providing thermal neutron removal between adjacent assemblies. Flux traps are regions of water separated by neutron absorber panels - neutrons pass through one panel, get thermalized in the interstitial water region, and then cannot get back through the thermal neutron absorber panels to induce fission events in the fuel material. This particular system was chosen to evaluate the Sourcerer capability because it represents a very challenging problem in terms of fission source convergence. The challenges arise from the reduced reactivity of the as-loaded UNF representation and the strong reduction in the thermal neutron transmission, which significantly decreases the coupling between the reactive regions (fissionable materials) in the system. These
particular source convergence challenges of detailed canister-specific analyses are not typically encountered in other criticality safety problems that assume uniform loading and fuel assemblies with much higher reactivity.

2 CODES, DATA, AND MODEL DESCRIPTION

The SCALE criticality code used for this evaluation is the KENO-VI three-dimensional MC eigenvalue computer code [6], using the continuous-energy ENDF-VII library.

The SCALE model for the TSC was created during development of the Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) [5]. Figure 1 shows a cut-away view of the SCALE model used in this analysis with some mixtures and surfaces omitted for visualization of the internal basket configuration. The UNF canister has an outer diameter of 170.33 cm and a height of 474.5 cm, consistent with the NAC-UMS Class 1 system [5]. The canister system is represented as fully flooded and reflected by 50 cm of water in the radial direction and from the top and bottom, to address criticality safety during UNF transportation per 10 CFR 71.55 requirements [7].

Figure 1. Cut-away view of TSC SCALE model.
Fig 2 shows a radial cross-section image of the canister contents listing the initial enrichment, burnup, and discharge date. Each fuel assembly was represented with 18 axial nodes for a total of 432 unique nodal isotopic concentrations to describe the UNF. The UNF isotopic compositions used in this model were calculated using SCALE/ORIGEN [8] via UNF-ST&DARDS.

3 CONVERGENCE OF EIGENVALUE MONTE CARLO WITH UNIFORM STARTING SOURCE DISTRIBUTION

The standard power iteration method, used in most MC criticality codes, involves starting with a “guessed” initial source distribution, and then iterating and updating the fission source until it is sufficiently converged. Traditionally, the initial source distribution is specified by the user or by default in the MC code. For example, the default in KENO is to start neutrons uniformly throughout all regions containing fissionable material. Since the initial “guessed” distribution does not usually represent the true (fundamental) fission source distribution, MC practitioners specify a number of inactive cycles, often referred to as “skipped cycles,” to be discarded before accumulating $k_{eff}$ in the “active cycles” [6]. The number of active cycles, which is primarily determined by the convergence criterion (i.e. target uncertainty), also depends on the difficulty of the source convergence that depends on the dominance ratio [9] of the problem. MC problems with dominance ratios close to 1 require more skipped and active cycles to achieve source convergence [9]. Furthermore, a bias in the calculated $k_{eff}$ is possible if the number of...
neutrons per cycle is too small. Therefore, it is crucial to ensure that a sufficient number of neutrons is simulated in each cycle for the bias in $k_{\text{eff}}$ to become negligible [9]. However, increasing the number of neutrons per cycle decreases the total number of cycles, for a fixed simulation time, which increases the statistical uncertainty and decreases the reliability of the calculated $k_{\text{eff}}$.

Users of MC codes for eigenvalue calculations must define the initial source distribution and several problem-dependent parameters (i.e., number of cycle, number of histories per cycle, and number of skipped cycles) that directly impact the convergence of the eigenvalue. Experienced users define the initial source distribution and parameters based on problem characteristics and past experience, and subsequently judge the appropriateness of their selections on diagnostic information that the codes produce. Although it is well known that poor starting source and parameter selection can lead to an under-estimation of the eigenvalue - which is a very troublesome possibility in the field of criticality safety - relatively little guidance is available to new users related to proper selection of the starting source and these fundamental, problem-dependent parameters [9]. In this work, the effects of changing these crucial parameters on the efficiency and reliability of MC eigenvalue calculations were investigated for a canister-specific UNF criticality safety problem.

To facilitate canister-specific $k_{\text{eff}}$ calculations as a function of time for a large number of loaded UNF canister systems, the generation of input files, execution of calculations, and output file processing are being automated within UNF-ST&DARDS. To ensure reliability and efficiency in the corresponding large number of calculations it is important to first determine the appropriate balance between the number of neutrons per cycle and the numbers of skipped and active cycles that should be used for this problem. To support this determination for the TSC problem, the numbers of neutrons per cycle and numbers of skipped and active cycles were varied for an equal number of total simulated neutrons. The number of skipped cycles was set to be one-third of the total number of cycles while the number of neutrons per cycle was varied between 100 and 500,000. The total number of simulated neutrons in the active and the skipped cycles was approximately $2\times10^7$ and the computational time was limited to 12 hr for each of the KENO MC calculations using the Oak Ridge National Laboratory Linux cluster cpile2. For this analysis, constraining the computational time of these calculations was essential to replicate the realistic conditions of canister-specific criticality analysis because numerous MC eigenvalue calculations need to be performed and repeated for the different compositions and configurations of each specific canister within reasonable running time. To keep the total number of neutrons approximately constant, the number of skipped cycles and the number of active cycles were changed with the number of neutrons per cycle. Figure 3 shows the $k_{\text{eff}}$ calculated using these different variations of KENO parameters; all calculations used a starting source uniformly distributed in all fissionable regions.

For validation purposes, a reference $k_{\text{eff}}$ value was determined based on 10 independent KENO calculations with different random seeds and 100,000 neutrons per cycle, 500 skipped cycles, and 1000 active cycles. The computational time for each of the independent calculations was in the 3.5 - 3.8 day range. The reference $k_{\text{eff}}$ (0.689451 ± 0.000026), also shown in Fig. 3, was calculated by averaging the results from the 10 independent calculations.
For KENO calculations with a fixed run time, the numbers of skipped and active cycles are dependent on the number of simulated neutrons per cycle. Even though increasing the number of neutrons per cycle enhances the sampling adequacy, the resulting decrease in the number of skipped and active cycles causes the $k_{\text{eff}}$ to be too far from the reference $k_{\text{eff}}$ when the number of neutrons per cycle exceeds 60,000. The rest of the analysis in this work used 30,000 neutrons per cycle and 500 active cycles which correspond to the highest number of neutrons per cycle that produced a $k_{\text{eff}}$ within $3\sigma$ from the reference $k_{\text{eff}}$. The run time for the 500 active cycles with 30,000 neutrons per cycle was approximately 6 hr.

An eigenvalue produced using an insufficiently converged fission distribution is not reliably computed and should not be used in criticality safety analysis [1, 9]. Therefore, it is necessary to assess the convergence of both $k_{\text{eff}}$ and the fission distribution to determine the number of cycles that should be skipped in a reliable MC calculation of $k_{\text{eff}}$ [9]. It has been demonstrated that the convergence of the fission distribution can be characterized using Shannon entropy, which converges to a single steady-state value as the fission distribution becomes stationary [10]. In the analysis shown in Fig. 3, the choice of skipping one-third of the cycles was arbitrary. This does not guarantee that the number of skipped cycles is appropriate for a reliable $k_{\text{eff}}$ calculation. Analyzing the convergence behavior of both $k_{\text{eff}}$ and the Shannon entropy will be demonstrated for different initial sources in Sect. 4.3.

### 4 HYBRID APPROACH IN SCALE FOR MONTE CARLO FISSION SOURCE CONVERGENCE

Within the SCALE code system, the automated source convergence sequence (Sourcerer) was developed to provide KENO [5] with initial starting sources based on information from relatively fast, approximate deterministic neutron transport calculations using the discrete
Hybrid technique in SCALE for fission source convergence applied to used nuclear fuel analysis

ordinates code Denovo [11]. To automate the coupling between the Denovo and KENO codes, Sourcerer automatically maps the KENO geometry on a user-defined grid, runs the Denovo eigenvalue deterministic calculation, and then processes the Denovo output to create the starting fission source, \( s(\vec{r}, E) \) for the KENO calculations. Using the fluxes calculated by an eigenvalue Denovo calculation, \( s(\vec{r}, E) \) is calculated according to

\[
s(\vec{r}, E) = \chi(E) \int_{E' = 0}^{\infty} u(E') \Sigma_f(\vec{r}, E') \phi(\vec{r}, E') dE',
\]

where \( \chi(E) \) is the fission spectrum, \( u(E) \) is the average neutrons emitted per fission, \( \Sigma_f(\vec{r}, E) \) is the fission cross section, and \( \phi(\vec{r}, E) \) is the energy- and space-dependent neutron flux.

Because the deterministic calculations are only needed to calculate an approximate fission source, the meshes used with this hybrid technique can be relatively coarse to speed up the computational efforts of the deterministic eigenvalue calculations. To enhance the fidelity of the deterministic models that typically use relatively coarse meshes, the macromaterials approach can be used in Sourcerer. The macromaterials approach uses a volume-weighted average of the materials in the MC model that exist in the space spanned by each element of the deterministic mesh [12]. Because of the importance of capturing the differences in the materials compositions between the different regions in cask-specific criticality safety analysis, the original macromaterials approach that used a point-testing method for defining the deterministic model materials was modified using to use a ray-tracing technique. Unlike the point-testing method, which samples the materials using a series of test points to determine the macromaterial fractions for each element in the deterministic mesh, the ray-tracing method shoots rays from all three directions of the mesh boundaries and determines the material fractions along the path of the ray until it leaves the mesh.

4.1 Methodology

For assessing the efficiency and reliability of the new approach, Sourcerer was used to deterministically calculate a starting source for the KENO eigenvalue calculations of the TSC problem. A uniform mesh with a square base having a 5 cm side length and a 10 cm height was used, corresponding to a total of 166,212 mesh elements. Because the only goal of the deterministic calculation is to provide an approximate starting source for the MC calculation, one can use meshes that are considered too coarse for eigenvalue deterministic calculations. The deterministic calculations used an 8 neutron group ENDF-VII library. With a tolerance of \( 10^{-5} \) in the Krylov iterations and a tolerance of \( 10^{-6} \) in the outer eigenvalue iterations, the running time of the Denovo eigenvalue calculation was 5.0 hr. To evaluate the impact of these convergence tolerances, an alternate case with a \( 10^{-3} \) tolerance for both the eigenvalue and the transport sweep iterations was also considered. The run time of the Denovo eigenvalue calculation of the alternate case was 1.0 hr. Both the calculations with the tight and the loose tolerances used a \( P_0 \) scattering order and a quadruple range quadrature set of two polar angles and two azimuthal angles. Changing and/or increasing the angular discretization were not found to significantly affect the calculated fission source distribution in the canister. Increasing the scattering order to \( P_3 \) increased the running time by a factor of 7.
4.2 Reliability of $k_{\text{eff}}$ calculations

KENO calculations with 30,000 neutrons per cycle and 500 active cycles were run with different random number seeds and different numbers of skipped cycles using: a uniform starting source; the starting source calculated deterministically with the loose tolerances; and the starting source calculated deterministically with the tight tolerances. For each fixed number of skipped cycles, 100 independent replications with a different random number seed were used to assess the reliability of the MC $k_{\text{eff}}$ calculation. Figure 4 shows the probability of “not obtaining” a $k_{\text{eff}}$ value that agrees with the reference $k_{\text{eff}}$ value within $3\sigma$ (~75 pcm).

![Figure 4](image)

**Figure 4.** Frequency of failure to agree with the reference $k_{\text{eff}}$ value for KENO calculations with different starting sources.

With a uniform starting source, the number of independent KENO calculations that agreed with the reference $k_{\text{eff}}$ within $3\sigma$ was still more than 5% until 350 cycles or more were skipped. With the deterministically calculated starting sources, $k_{\text{eff}}$ agreed with the reference $k_{\text{eff}}$ calculation within $3\sigma$ for all the KENO calculations when 50 or more cycles were skipped. This demonstrates that the use of deterministically calculated starting sources causes a significant improvement in the reliability of MC $k_{\text{eff}}$ calculations for a given analysis runtime.

4.3 Efficiency of $k_{\text{eff}}$ calculations

The variations of the Shannon entropy and $k_{\text{eff}}$ as functions of the cycle number along with the reference $k_{\text{eff}}$ value are shown in Fig. 5 for the TSC problem using a uniform starting source, a starting source calculated deterministically with loose tolerances, and a starting source calculated deterministically with tight tolerances. All the KENO calculations used 30,000 neutrons per cycle except for the reference $k_{\text{eff}}$ calculation, which is described in Sect. 3. Both the $k_{\text{eff}}$ and the Shannon entropy plots were examined for the efficiency comparison, but the Shannon entropy plots solely determined the number of skipped cycles because the Shannon entropy converged more slowly than $k_{\text{eff}}$ for all cases. It is worth mentioning that this work only focuses
on calculating $k_{eff}$ in canister-specific UNF problems and does not investigate other quantities such as local interaction rates or power. In this analysis, the only goal of examining the convergence of the fission distribution via the Shannon entropy is to ensure that $k_{eff}$ is not accumulated before the fission distribution becomes stationary, and therefore does not falsely converge due to undersampling of important reactive regions [1, 9].

![Figure 5. Entropy and $k_{eff}$ variations for TSC with 30,000 neutrons/cycle.](image)

The number of skipped cycles was determined from the number of cycles after which the entropy falls inside a bandwidth determined by the average and the population standard deviation of the entropy of 1000 cycles that start after the systematic change ceases to be visually apparent. These 1000 asymptotic cycles were counted after 750 cycles for the uniform starting source and 200 cycles for the deterministic starting sources with the loose and the tight tolerances. The circles superimposed on the Shannon entropy graphs in Fig. 5 show these numbers of skipped cycles for each starting source case. These numbers of skipped cycles were used in subsequent $k_{eff}$ calculations that were used to calculate the efficiency enhancements provided by Sourcerer.

After determining the number of skipped cycles, the $k_{eff}$ calculations were repeated using the uncertainty threshold ($1\sigma = 0.00025$) to terminate the KENO execution. All the $k_{eff}$ calculations used 30,000 neutrons per cycle and the numbers of skipped cycles determined using the Shannon entropy plot of Fig. 5. The parameters used in these KENO calculations are shown in Table I. In addition to decreasing the number of skipped cycles, the use of deterministically calculated starting sources was found to also decrease the number of active cycles required to reach the uncertainty threshold because of decreasing the cycle to cycle variation of $k_{eff}$ in this problem.
Table I. Calculation parameters for the TSC problem

<table>
<thead>
<tr>
<th>Starting source</th>
<th>Cycles</th>
<th>Time (min)</th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Skip</td>
<td>Active</td>
<td>Total</td>
<td>Denovo</td>
<td>KENO</td>
</tr>
<tr>
<td>Uniform</td>
<td></td>
<td>439</td>
<td>481</td>
<td>920</td>
<td>0.00</td>
<td>719.0</td>
</tr>
<tr>
<td>Loose tolerances deterministic</td>
<td></td>
<td>45</td>
<td>442</td>
<td>487</td>
<td>68.7</td>
<td>350.6</td>
</tr>
<tr>
<td>Tight tolerances deterministic</td>
<td></td>
<td>40</td>
<td>399</td>
<td>439</td>
<td>233.4</td>
<td>304.9</td>
</tr>
</tbody>
</table>

The efficiency improvements caused by using deterministic starting sources were measured by comparing the time needed by the MC calculation to achieve the uncertainty threshold using a uniform starting source to the total time (deterministic + MC) needed to reach the same uncertainty threshold using deterministic starting sources. The results of the Sourcerer/KENO calculations for the TSC problem are shown in Table II for the different starting sources. All the calculated $k_{\text{eff}}$ agreed with the reference $k_{\text{eff}}$ within $2\sigma$ (~50 pcm).

Table II. $k_{\text{eff}}$ and speedup comparison for the TSC problem

<table>
<thead>
<tr>
<th>Starting source</th>
<th>$k_{\text{eff}}$</th>
<th>Speedup</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uniform</td>
<td>$0.68977 \pm 0.00025$</td>
<td>1.00</td>
</tr>
<tr>
<td>Loose tolerances deterministic</td>
<td>$0.68944 \pm 0.00024$</td>
<td>1.71</td>
</tr>
<tr>
<td>Tight tolerances deterministic</td>
<td>$0.68900 \pm 0.00024$</td>
<td>1.36</td>
</tr>
</tbody>
</table>

The use of a deterministic starting source provided a factor of 1.71 speedup in the convergence of the $k_{\text{eff}}$ calculation. Tightening the tolerances decreased the speedup gain by 20% because of the extra running time required by the Denovo eigenvalue calculation, but was still 36% more efficient than using a uniform starting source distribution. For this problem, increasing the angular discretization and/or increasing the scattering order of the Denovo eigenvalue calculation was found to significantly increase the time requirements of the Denovo calculation without affecting the convergence of the KENO eigenvalue calculation.

Figure 6 shows the calculated $k_{\text{eff}}$ with 500 active cycles and 30,000 neutrons per cycle as a function of the number of skipped cycles for the KENO calculations with the uniform starting source and the two deterministically calculated starting sources with tight and loose tolerances.

The KENO calculations with a uniform starting source started to agree with the reference $k_{\text{eff}}$ within $3\sigma$ (67 pcm) after skipping 225 cycles and within $2\sigma$ (47 pcm) after 400 cycles. Even after these agreements, the calculated $k_{\text{eff}}$ with a uniform starting source was systematically lower than the reference $k_{\text{eff}}$ for all cases that spanned 10 to 500 skipped cycles. With deterministically calculated starting sources, the $k_{\text{eff}}$ started to agree with the reference $k_{\text{eff}}$ within $2\sigma$ (~45 pcm) after only 10 cycles. The expected statistical fluctuation around the reference value was also observed for the KENO calculations with the deterministic starting sources. The first $k_{\text{eff}}$ value that was higher than the reference $k_{\text{eff}}$ appeared at 75 cycles with a starting source calculated deterministically with loose tolerances and appeared at 50 cycles with a starting source calculated with the tight tolerances deterministic calculation.
Figure 6. $k_{eff}$ as a function of the number of skipped cycles with 500 active cycles and 30,000 neutrons/cycle.

5 CONCLUSION

The combination of large variations in assembly reactivity, low-reactivity assemblies, and the presence of strong neutron absorbers between assemblies can cause difficulties with source convergence in criticality safety analyses of actual, as-loaded UNF canisters. To evaluate its effectiveness for addressing these difficulties, the Sourcerer sequence that automates the coupling between deterministic and MC eigenvalue calculations has been applied to the TSC problem. Sourcerer uses a relatively fast, approximate deterministic transport calculation to calculate a starting fission source for an MC eigenvalue calculation. The use of Sourcerer takes the guesswork out of defining an appropriate, problem-dependent starting source. Additionally, the more accurate starting source provided by the deterministic calculation decreases the probability of producing inaccurate tally estimates associated with undersampling problems caused by inadequate source convergence. Furthermore, the use of Sourcerer was found to increase the efficiency of the overall simulation, even with the inclusion of the extra computational time required by the deterministic calculation, by reducing the number of skipped cycles.

This work complements ongoing activities to standardize/modularize the use of the new sequence in all UNF cask-specific analysis. Further parametric analyses are ongoing to evaluate the sensitivity of Sourcerer’s efficiency and reliability enhancements with the variations in the Monte Carlo and the deterministic parameters.

6 ACKNOWLEDGMENTS

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UNF canister problem was sponsored by the Department of Energy Office of Nuclear Energy Nuclear Fuels Storage and Transportation Planning Project.

7 REFERENCES


