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Hypothetical Sodium-Cooled Fast Reactor Fuel Transport Using the Existing ES-3100

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Nuclear Energy and Fuel Cycle Division

HYPOTHETICAL SODIUM-COOLED FAST REACTOR FUEL TRANSPORT USING THE EXISTING ES-3100

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1. INTRODUCTION

Increased industry interest in increased enrichment fuels is associated with a heightened interest in high-assay low-enriched uranium (HALEU)-based systems, as novel designs look to take root as alternatives to traditional LWRs. Increased enrichment with novel reactors can produce designs that, in theory and in some historical experience, are capable of operation at increased burnups, higher energy density, and other unique features compared to conventional LWRs. As part of a Department of Energy (DOE) initiative to increase the availability of HALEU fuel, initial funding sourced from the Inflation Reduction Act of 2022 (H.R. 5376) [1] resulted in the DOE/NRC Criticality Safety for Commercial-Scale HALEU Fuel Cycle and Transportation (DNCSH) project, part of the HALEU Availability Program. The project aims to support the Nuclear Regulatory Commission (NRC) in providing data for criticality safety validation of reactor designs that, while perhaps demonstrated in limited capacity, represent more exotic systems than those that regulators are accustomed to reviewing.

As part of the DNCSH project, an initial survey of HALEU fuel forms under development in the Advanced Reactor Demonstration Program and transportation packages licensed for HALEU fuel transport was performed and presented in part to the public in a February 2024 workshop [2, 3]. The slides for the presentation are provided online by the NRC, and a summary of the workshop's relevant discussion and community participation is available [3, 4]. This presentation included the analysis of two hypothetical transportation scenarios for HALEU-based systems, determining a validation basis for the scenarios with existing critical experiments. This report details the analysis of one of these scenarios, representative sodium fast reactor (SFR) fuel placed into an existing ES-3100 transportation package [5]. A sister report details a similar hypothetical analysis regarding tri-structural isotropic (TRISO) particle fuel pebbles placed into a conceptual large-volume transport package [6].

In this report, Section 2 provides a description of the model used in this effort, as well as design choices in consideration of the package and SFR fuel. Section 3.1 details the calculations performed to quantify the eigenvalue of the SFR fuel within the ES-3100 so as to determine the subcriticality of the system. Section 0 uses sensitivity and uncertainty (S/U) analyses to generate sensitivities and nuclear data–induced uncertainties for use in identifying experiments for validation. Section 4 provides concluding remarks on the findings of the S/U-based approach for validation of the SFR fuel in the ES-3100.

1.1 PACKAGE BACKGROUND

The package chosen for this work is an existing design developed by Y-12 for the transport of highly enriched uranium (HEU). The ES-3100 is a 30-gal drum-type package, with a containment vessel and confinement assembly to protect fissile material during transport and to maintain structural integrity [5]. The package largely consists of the steel used to form these boundaries and a solid fill material to prevent damage from physical deformation as well as to act as insulation from thermal stress. It is approved for HEU shipments on the order of tens of kilograms, depending on material form and composition [7]. The ES-3100 safety analysis report for packaging (SARP) schematic is reproduced on the left side of Figure 1, showcasing the inner and outer boundaries and fill material [5].

1.2 FUEL BACKGROUND

For the SFR fuel, the reference 250 MWth Advanced Burner Test Reactor (ABTR) metallic fuel was chosen as a representative model, having been described in a public benchmark by Argonne National Laboratory, with analysis in several public Oak Ridge National Laboratory reports [8-10]. The reactor design consists of fuel assemblies containing a mixture of transuranics (TRU) and plutonium alloyed with zirconium, intended to fission the waste products and actinide material. The fuel rod shown on the right of Figure 1 is nearly 3.5 m tall during operation with a fuel region of 84 cm, or 80 cm before thermal expansion. The fissile region is the region of most interest and relevance for criticality safety regulatory concerns. The analysis in this report, with no knowledge of industry plans for transporting SFR fuel, assumes that the fissile material would be shipped in the form of fuel slugs, packaged following initial fabrication, with final assembly into a fuel rod performed elsewhere (perhaps on the reactor site) into the cladding with the reflector and sodium bond. Thus, the ES-3100 can be used as a hypothetical transport package, as the manufactured fuel slug physically fits into a minimally modified version of the existing package design.

In ensuring the safe transport of fissile material, an applicant must perform a criticality evaluation of the package and its contents, and the NRC must perform a review of the evaluation with independent confirmation of the package's suitability. Given that the technological readiness of HALEU-based systems is continually advancing, and that multiple construction permits of HALEU-based reactors have been submitted, the NRC must be prepared with the tools, retained knowledge, and technical basis for performing independent criticality analyses of HALEU-based systems. Tool readiness has been demonstrated [9,10], whereas DNCSH aims to expand the technical basis to support criticality evaluations, primarily through the necessary aspect of validation. To assess the status of the available critical experiments in support of validation, the work documented in this report was performed as a measure of critical experiment applicability for 20 wt% ^{235}U metallic SFR fuel slugs placed within the ES-3100.

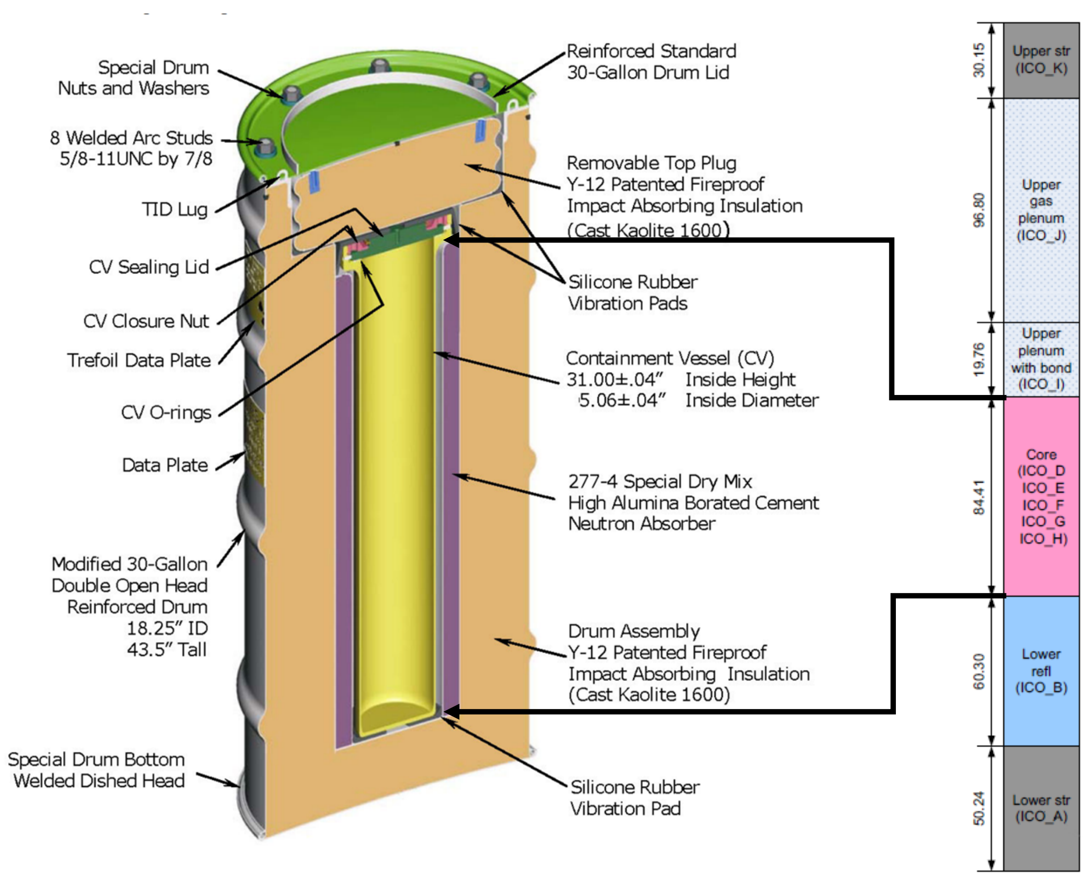

Figure 1. ES-3100 schematic for HEU transport [5] (left) and ABTR fuel rod design at operating conditions [8] (right).

2. MODEL DESCRIPTION

Although the ES-3100 canister design was taken from multiple sources, the primary source of the information is the SARP [5]. The major design characteristics were preserved, and a rendering generated with SCALE 6.3.1 is provided in Figure 2 [11]. A containment vessel, consisting of a nested capped steel cylinder within a larger steel cylinder was modeled as shown in Figure 2. General specifications of the ES-3100 call for a containment vessel with inner dimensions of 31 in. in height and 5.06 in. diameter (78.74 and 12.85 cm, respectively) [5]. One adjustment was made to this dimension to account for the length of the ABTR fuel. The fuel height of a fuel slug in the ABTR is 80 cm [8]; therefore, the height of the containment vessel was increased by 1.26 cm to account for this slight difference. The effect of this modification on k_{eff} was statistically negligible, and there is sufficient space in the drum to accommodate this minor increase in containment vessel height. The drum was modeled with an inner diameter of 48.68 cm and a height of 110 cm. This is slightly different than the SARP dimensions, due simply to initial conflicting dimensions of the 30-gal drum from different sources. Resolving the discrepancy of 1-2 cm variation in width was not considered a concern for significantly impacting the calculations in this study. Specifics of the package, such as top plugs, nuts, washers, etc., were not relevant or immediately available for modeling in detail. None of the borated cement absorber was credited.

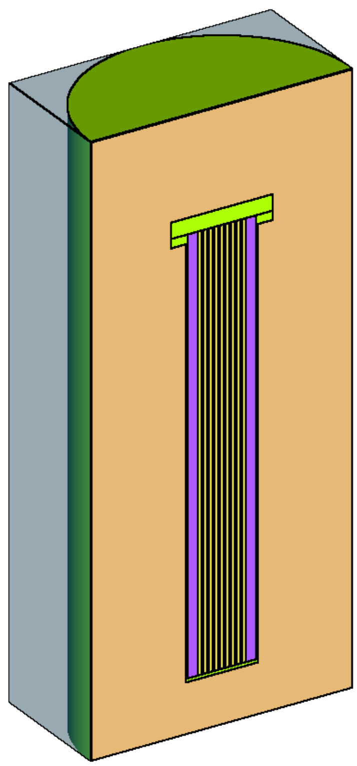

Figure 2. SCALE model of ES-3100.

Several fill materials were considered for the drum. Kaolite 1600™ (a mixture of Portland cement, water, and vermiculite) is the nominal fill material of the ES-3100. Water was modeled to consider water intrusion. The material definition of Kaolite was determined from a *Thermal Ceramics* specification sheet included in the first revision of the SARP [12]. Two Kaolite materials were studied upon determining a chemical composition. The tested material definitions were not different by chemical weight compositions—rather, the densities were adjusted to the upper (0.625 g/cm^3) and SARP (0.359 g/cm^3) estimates of the density [5]. Kaolite is also liable to high moisture content. However, the Kaolite in either condition was inevitably determined to be the least reactive condition by a substantial margin, so for the purposes of this analysis, no further study was pursued. Water and stainless-steel 304 compositions were defined using SCALE standard composition (StdComp) definitions [11]. All materials were modeled at room temperature (293 K).

Table 1 lists the examined scenarios and differences in fill material between them, with the differences also shown in Figure 3. Fuel is shown in yellow; steel in green; water in blue; void in purple; and Kaolite in orange. Two scenarios were investigated: the ES-3100 with no in-leakage but in an infinite submerged array (Dry); and the ES-3100 with in-leakage into the canister drum and containment vessel, in an infinite submerged array (Wet). Per the SARP, "*Testing conducted in accordance with the physical testing requirements of 10 CFR 71 demonstrated that water leakage into the containment is not a credible event under the NCT and HAC,*" but flooding was considered "*per 10 CFR 71.55(b)*" [5]. Inputs and other generated data are available on the DNCSH Gitlab repository[1].

Table 1. Conditions of investigated scenarios.

Scenario	**Dry**	**Wet**
Containment Vessel	Dry (Void)	Flooded (Water)
Canister Drum	Dry (Kaolite)	Flooded (Water)
Inter-Canister	Submerged (Water)	Submerged (Water)
Number of Canisters	Infinite Array	Infinite Array

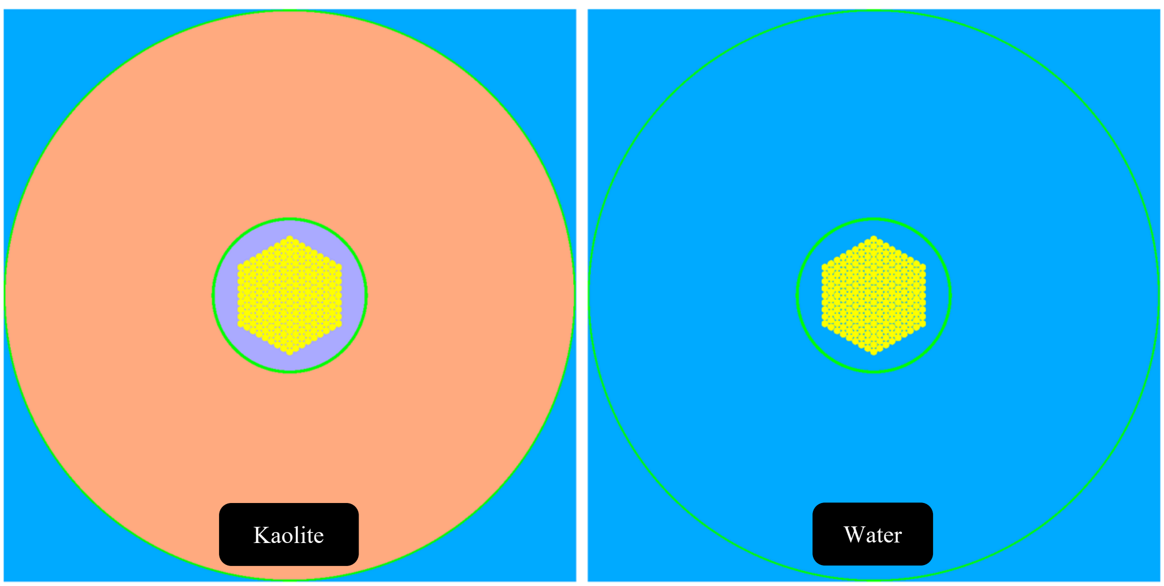

Figure 3. Rendering of dry (left) and wet (right) configurations.

For fuel placed within the containment vessel, the ABTR was the source of fuel dimensions for a representative SFR. Although it is intended to operate with TRU and plutonium on startup, as a representative reactor, the dimensions were maintained while the fuel composition was markedly simplified to consist of 20 wt% ^{235}U in a U-10Zr metal form. The density was modeled at 16.1 g/cm^3. The slug retained the dimension and form of the 250 MWth ABTR fuel [8], but the fissile inventory was simplified to remain within the scope of the project (HALEU, not TRU). The slug radius was 0.3014 cm, with a height of 80 cm [8]. The initial configuration of slugs modeled a close-packed hexagonal array with a triangular pitch equal to the fuel radius. This arrangement represents the tightest configuration of fuel; gaps between slugs are difficult to visualize in Figure 3. A single assembly's worth of fuel (217

[1] https://code.ornl.gov/dncsh/applications.

slugs), approximately 14.3 kg of ^{235}U, can fit into the containment vessel. There is space and mass allowance for additional fuel, but there is insufficient space in the containment vessel as modeled for an additional assembly's worth of fuel slugs. Logistically, integer assembly quantities of fuel per package were considered practical, so the container was modeled with 217 fuel slugs. The containment vessel could be further expanded—as was done to account for the vertical clearance of the fuel—to account for an additional assembly, but the consideration of such a change in design was considered too significant for this study in deviating from the true design.

3. RESULTS

SCALE/CSAS results were generated for the SFR fuel slugs in the ES-3100 for various configurations. Primarily, the fill of the canister drum was altered to consider different materials, and the array pitch of fuel in the containment vessel was adjusted to consider fuel shifting given the absence of any credited spacers. SCALE/TSUNAMI results were generated for similar configurations, excluding the intermediate array pitch adjustments. Only the nominal and most expanded array pitches were individually investigated. CSAS results had final stochastic uncertainties of 10 pcm, and TSUNAMI results had final stochastic uncertainties of 15 pcm. Continuous-energy ENDF/B-VII.1 neutron cross section libraries were used for both CSAS and TSUNAMI sequences. For TSUNAMI sensitivity generation, the iterated fission probability (IFP) method with five latent generations was used.

3.1 CSAS EIGENVALUES

Table 2 presents the basic configurations and their calculated eigenvalues. All k_{eff} results were below 0.6 for a single assembly's worth of 20 wt% fuel slugs (14.3 kg ^{235}U in total). With the close-packed array of HALEU metal pins, even flooded and reflected with water in an infinite array, the spectrum—as measured by the energy of average lethargy causing fission (EALF)—is intermediate. Given the enrichment, fuel form, and spectrum, this would be categorized as an "IEU-MET-INTER" (IMI) system, representing an intermediate enriched uranium, metal, intermediate energy system, using the International Criticality Safety Benchmark Evaluation Project (ICSBEP) categorization of critical experiments [13]. The full extent of configuration differences is outlined in Table 1. In general, the wet condition is considered the most realistic limiting configuration, so attention is preferentially given to it. The Kaolite-filled configuration is the least reactive.

Table 2. Calculated eigenvalue by evaluated configuration.

Scenario	Dry	Wet
k_{eff}	0.51690	0.58452
EALF (keV)	0.355	0.137

Recognizing the importance of moderation and acknowledging that a close-packed array is unstable without spacers and likely undermoderated, the pitch of the fuel slugs was increased to the geometrical constraints of the containment vessel, in 0.02 cm increments. The largest array pitch of 0.3814 cm is pictured in Figure 4, representing the loosest packing of the array. The results of this increased moderation in the wet configuration are presented in Table 3 with calculated k_{eff} and the EALF. Each increment increased reactivity by 0.024–0.031 in Δk_{eff}, increasing in incremental worth with additional moderation. While an unstructured array of slugs may be more reactive than this maximum ordered pitch configuration, there are physical limitations on potential arrangements given the limited spacing within the containment vessel and the margin to criticality remains substantial.

Table 3. Increase in k_{eff} with increased slug pitch.

Slug Pitch (cm)	0.3014	0.3214	0.3414	0.3614	0.3814
k_{eff}	0.58452	0.60935	0.63632	0.66432	0.69454
EALF (keV)	0.137	0.093	0.063	0.042	0.028

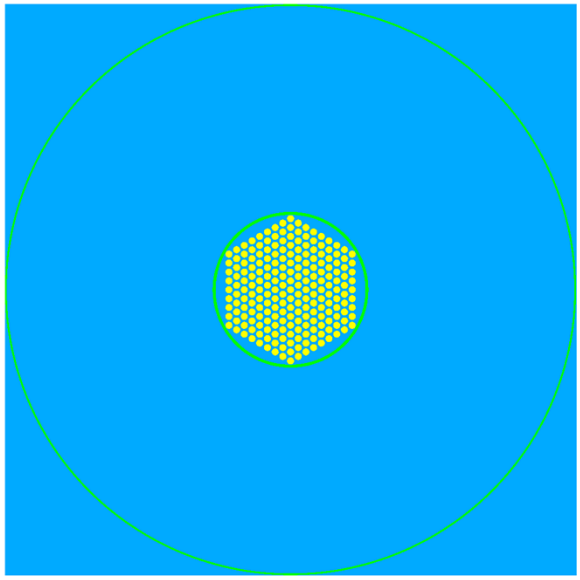

Figure 4. Example of expanded slug pitch (0.3814 cm).

The density of the Kaolite was perturbed to determine whether the upper or lower bounding density was more reactive. The specification sheet provides a density range of 320–625 kg/m^3, depending on the method of application (casting or gunning) [12]. The density evaluated by the SARP was 0.359 g/cm^3—not the lowest possible density but sufficiently low, and it is a realistic representation of the ES-3100 casting fill in practice. This resulted in a k_{eff} of 0.53394 for a density of 0.359 g/cm^3, and a k_{eff} of 0.51690 for a density of 0.625 g/cm^3. The lower-density Kaolite is more thermal and reactive. This difference is insignificant relative to the wet, expanded pitch condition.

3.2 TSUNAMI SENSITIVITIES AND SIMILARITY ANALYSIS

TSUNAMI calculations were performed for three configurations: the close-packed wet and dry scenarios, as well as the most reactive, expanded pitch wet condition. The plots in Figure 5 and Figure 6 illustrate the difference in main nuclides of interest, ^{1}H, ^{235}U, and ^{238}U between the two wet conditions, demonstrating the influence of the additional spacing and moderation. ^{16}O has a significant sensitivity coefficient in both wet configurations, with an integrated sensitivity value of 0.11, with little energy dependent differences between the two. With increased moderation, the sensitivity to the primary moderator ^{1}H increases as expected. As sensitivity to ^{1}H increased, integrated sensitivities to ^{235}U and ^{238}U decreased with the expanded pitch. The sensitivity to ^{238}U resonances is greater with the expanded pitch: this was reflected by both the increased ^{1}H sensitivities mirroring the ^{238}U resonances as well as the increased magnitude of the sensitivity at ^{238}U resonances themselves. With increased moderation, the negative ^{1}H sensitivity at thermal energies decreases in magnitude; while not pictured, this is a result of the sensitivity to elastic scattering increasing, as well as a lessened sensitivity to the capture reaction. Although minor, this change was observed in the corresponding moderation effect on ^{235}U: with more moderation and more thermal neutrons, ^{235}U is more sensitive to thermal neutrons, increasing the importance of a thermal neutron for ^{235}U while decreasing for ^{1}H.

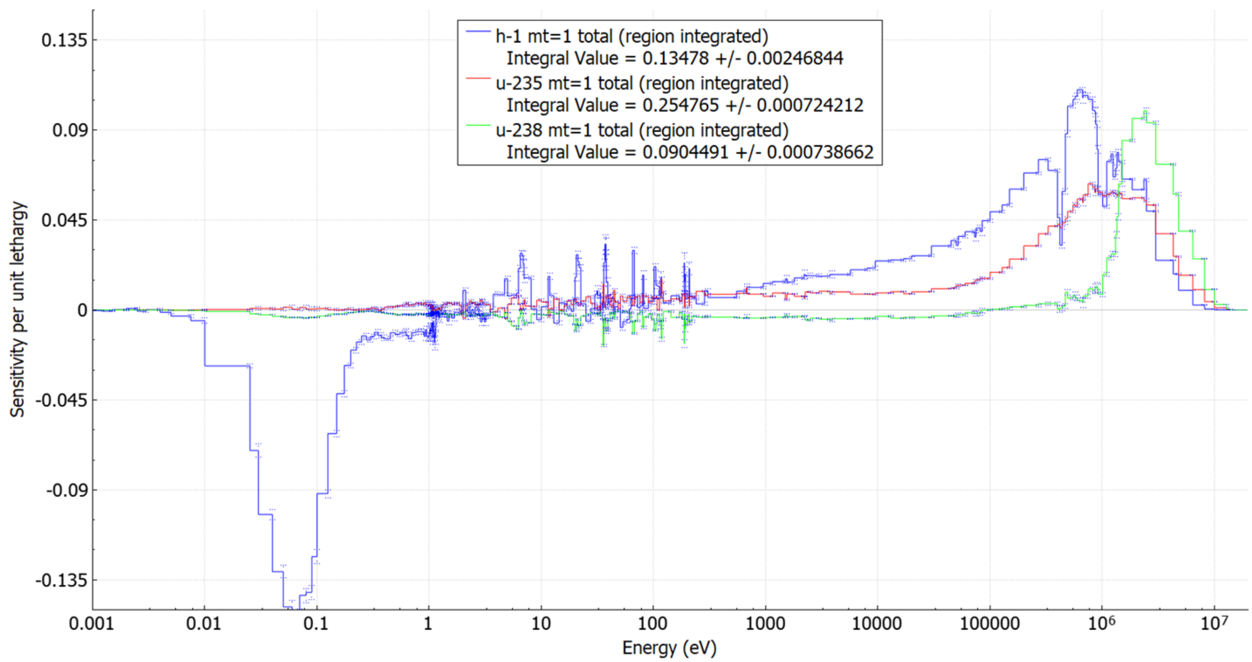

Figure 5. Sensitivity profile of the wet ES-3100 at the nominal pitch of 0.3014 cm.

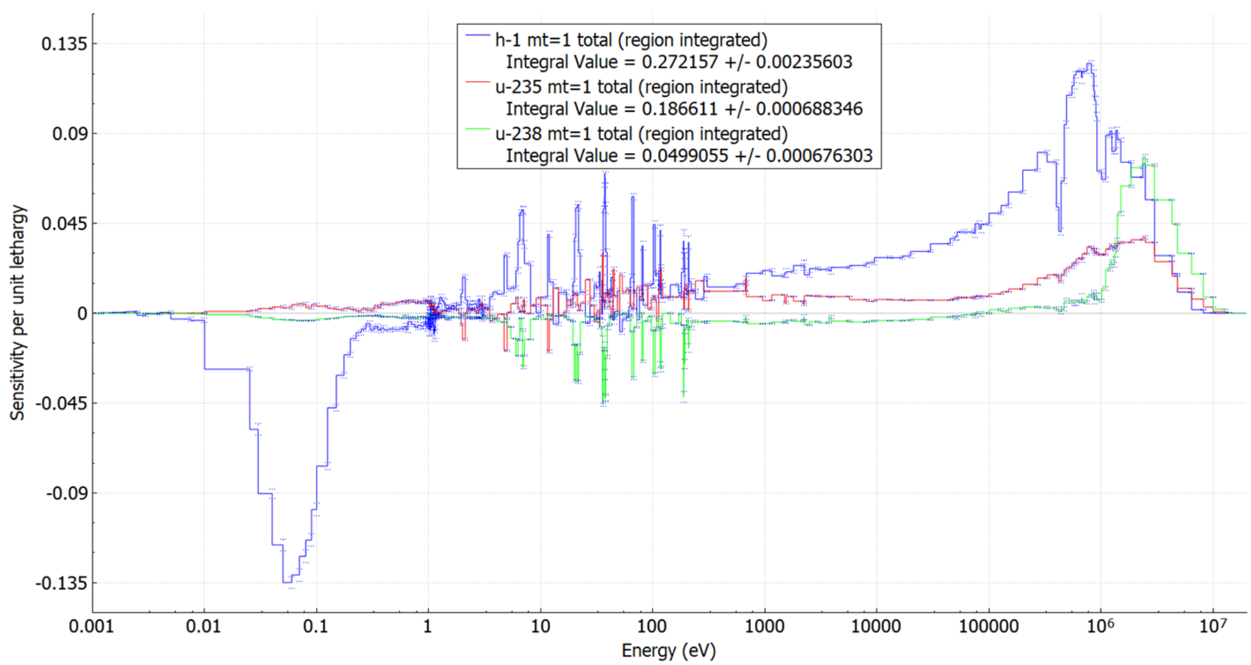

Figure 6. Sensitivity profile of the wet ES-3100 at the maximum pitch of 0.3814 cm.

With generated sensitivity data files (SDFs), the applications were examined against a swath of ICSBEP critical experiments using TSUNAMI-IP. SDFs used included those provided with the ICSBEP Handbook, as well as those in the ORNL Verified Archived Library of Inputs and Data (VALID) [13,14]. As noted above, the system characteristics would likely be categorized as an IMI experiment. There are a total of three cases in the ICSBEP-designated IMI experiments. With the expanded pitch, the additional thermalization could recategorize the designation as IMT (thermal spectrum) or IMM (mixed spectrum). No IMT configurations exist in the ICSBEP Handbook, whereas three IMM configurations are available, depending on the exact relation between EALF and the energy region in which a majority of fissions occur.

The generated sensitivity profiles, in conjunction with the ENDF/B-VII.1 56-group covariance library, included in SCALE 6.3.1, were used to produce estimated nuclear data–induced uncertainties in the eigenvalue—how each system is expected to respond to the uncertainties in the nuclear data used in the calculation. How similar the shared data-induced uncertainties between the applications and experiments is then determined with a correlation coefficient, c_k. c_k is one of several integral similarity indices considered valuable for S/U-based validation. The use of c_k provides a quantitative approach to selecting experiments beyond the more qualitative, yet informed, approach where system characteristics such as spectrum, fissile form, and enrichment, among other physical features, are used to determine suitable experiments. S/U-based identification allows for identification of experiments that may not conventionally be considered, by quantitatively determining the shared nuclear data-induced uncertainties between systems. Historical use of c_k as a selection criterion has supported the selection of applicable experiments for validation by choosing those with a c_k greater than or equal to 0.9 when compared with an application, indicating a high-degree of similarity [15]. Although not ideal, a c_k greater than or equal to 0.8, considered marginally similar, may also be used to demonstrate marginal applicability where high-similarity experiments do not exist [15].

Table 4 presents the total number of experiments meeting high and marginal c_k similarity thresholds as well as the maximum experiment c_k for the three configurations. Dry and wet scenarios do not have any experiments with a c_k at or above 0.8, meaning there are no available experiments considered even marginally applicable using S/U-based methods. This indicates that the current technical basis of experiments suitable to validate this hypothetical U-10Zr metal SFR fuel in the ES-3100 is insufficient, and the same conclusion may hold true broadly for HALEU metal SFR fuel. Further study could be performed to determine the cause, per se, of this shortcoming. For example, the fuel could be too subcritical to achieve high similarity to critical experiments due to an overall reduced sensitivity relative to critical cases. Alternatively, perhaps the intermediate energies have poor validation coverage, with misaligned sensitivities. These explanations are worthy of study, but they are not within the current scope of this analysis.

When considering the most reactive condition, the case of water ingress with two containment failures, complete loss of fill material, and spatial reconfiguration of the fuel to the most reactive array state, 373 experiments were found to be marginally similar. The maximum c_k in this state is 0.8947, and 91 of the 373 cases have a c_k greater than 0.85. Several hundred marginally applicable experiments may be considered sufficient for validation, particularly as several approach the 0.9, high-similarity, threshold. The number of experiments is likely sufficient given the subcriticality of the system; in a more reactive subcritical system nearing criticality, such a validation basis may be insufficient. Additionally, the variation in c_k due to the choice of covariance library has the potential to significantly affect the number of applicable experiments, though such a study on this system, in particular, is considered out of scope for this simple framing analysis [16]. Conversely, with such a significant margin to criticality even in the package's heavily compromised state, extensive validation for determining the most accurate bias and bias uncertainty to apply may be superfluous. While the expanded pitch configuration is a result of several mechanical failures and optimizations likely considered noncredible, the SFR fuel in the ES-3100 in this state likely has sufficient validation coverage.

Table 4. Calculated eigenvalue and c_k by evaluated configurations.

	Dry	**Wet**	**Wet (0.3814 cm)**
$\mathbf{k_{eff}}$	0.51690	0.58452	0.69454
$\mathbf{c_k \geq 0.8}$	0	0	373
$\mathbf{c_k \geq 0.9}$	0	0	0
Maximum $\mathbf{c_k}$	0.7236	0.7862	0.8947

Of the 373 marginally acceptable benchmarks for the most reactive configuration, 357 were LEU-COMP-THERM (LCT) experiments, 9 were LEU-MET-THERM (LMT), 2 were IEU-COMP-MIXED, and 5 were LEU-MISC-THERM. With the additional moderation provided by the expanded array, and EALF on the order of 20 eV, it is not surprising that many thermal benchmarks became applicable. Additionally, it is not surprising that even at an increased enrichment, LEU experiments are considered marginally applicable. Whereas 20 wt% ^{235}U is considered IEU by the ICSBEP categorization, it is typically considered LEU since many categorizations use only HEU and LEU designations. Such an arbitrary restriction may impact more conventional, judgement-based validation suites, while the use of S/U selection criteria identifies applicable experiments due to shared nuclear data-induced uncertainties alone. The risk of applying, or failing to apply, sufficiently similar experiments due to differences in nomenclature can be lessened.

Most surprising of the makeup of applicable experiments is the number of COMP, rather than MET experiments. Eighty-seven (87) LMT ICSBEP configurations exist, as opposed to 1559 LCT configurations, so a lesser number of applicable experiments is to be expected. Of the available 87 LMT experiments, 9 had a c_k greater than 0.8, with the highest being 0.8865. However, even as a fraction of available experiments, LCT experiments were more likely to be considered applicable, as 23% of LCT experiments in the ICSBEP had a c_k greater than 0.8, as opposed to 10% of LMT experiments. When considering that SFR designs exist with the use of oxide fuel, rather than metallic fuel, the ABTR being one such example, the use of oxide fuel may further increase the number of applicable experiments for similarly sized SFR fuel elements in the ES-3100.

4. CONCLUSIONS

The DNCSH project was established to assist the NRC in the criticality evaluation of new and upcoming fissile material package reviews. As increased enrichments and HALEU-based designs breakthrough into the commercial space, a sufficient technical basis must exist for the NRC to perform their reviews. Critical experiments with which to perform code and system validation must be applicable for these new systems. A range of enrichments, fuel forms, absorbers, and spectra, among other features, must be considered in assessing current readiness of the existing validation basis for advanced reactor fuel transportation. Each feature may introduce some deficiency in validation coverage, and plans for new critical experiments should be aware of these deficiencies to better support advanced reactor fuel validation.

On the analysis of SFR fuel in the ES-3100, the ABTR fuel design was modified to 20 wt% ^{235}U in U-10Zr and placed within the 30-gal drum-type package. Every configuration examined was well under subcritical limits in an infinite flooded array (below a keff of 0.7) when containing 217 slugs of SFR fuel, for a total of 14.3 kg ^{235}U. From a logistics and criticality perspective, the ES-3100 could prove useful as an existing package to use for the transportation of partial SFR fuel rods, following the fuel slug fabrication. The relatively compact package allows a full fuel assembly of fuel to be transported in a borderline standard 30-gal drum. With the 250 MW_{th} ABTR as the representative SFR, requiring 60 fuel assemblies, it is feasible to ship an entire reactor core of fuel slugs in one truckload and maintain subcriticality.

Of interest to the DNCSH project is the validation basis for the SFR fuel in the ES-3100. When considering the most reactive state, the SFR fuel in the ES-3100 was found to have a significant (373) number of marginally applicable experiments when the pitch of the fuel array was increased. While none had a c_k greater than or equal to 0.9, several were on the cusp of this threshold. Not noted in the table, ninety-one (91) experiments were at or above a c_k of 0.85. This indicates that a more moderated package of HALEU U-10Zr metal may have sufficient validation coverage. Although having critical experiments meeting the 0.9 c_k threshold is preferable, accounting for covariance matrix differences between libraries may render previously inapplicable experiments applicable strictly due to increased uncertainties attributed to nuclear data of moderate sensitivity. However, it is still preferable to have a validation suite of experiments with c_k values of 0.9 or greater, which has yet to be established for the surrogate SFR fuel in the ES-3100.

The instance of expanded pitch may be considered noncredible with containment failures, complete flooding, full reflection, fill ejection, and optimized array spacing in the same scenario. If so, and the limiting credible event is such as one described by the wet and dry scenarios, there would yet to be an established suite of experiments for the surrogate SFR fuel in the ES-3100. In both dry and wet configurations (both in an infinite flooded array), no experiments were found to be applicable or marginally applicable, with all c_k values below 0.8. A potential reason for this lack of applicable experiments could be the limited number of qualitatively similar experiments in the ICSBEP Handbook—that is, LEU/IEU experiments in a metal form with an intermediate spectrum. Additionally, in the nominal wet and dry conditions, the eigenvalue was substantially subcritical (k_{eff} below 0.6). Such a subcritical configuration may simply poorly reflect the spectral and sensitivity profiles of critical experiments. Regardless, there appears to be a gap in experimental applicability to HALEU U-10Zr metal in drum packages, particularly when under-moderated.

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