Transporting Existing VSC-24 Canisters Using a Risk-Based Licensing Approach

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1. Background

The VSC-24 Ventilated Storage Cask System is a canister-based dry cask storage system that is designed and licensed to provide safe storage of spent nuclear fuel (SNF) assemblies in accordance with Title 10 of the United States Code of Federal Regulations Part 72 (10CFR72). A brief description of the VSC-24 Multi-Assembly Sealed Basket (MSB), the canister for the VSC-24 System, is provided in Section 2.

The VSC-24 storage system has been implemented at three different nuclear plants in the United States: Arkansas Nuclear One (ANO), Palisades, and Point Beach nuclear plants. There are 58 VSC-24 casks loaded at these three sites. Currently there are no plans to deliver or load any additional MSBs at any sites in the United States. Eventually, the existing loaded MSBs will either have to be transported intact to a long-term storage facility or repository or opened at the plant to repackage the SNF assemblies for transportation. Transporting intact MSBs has clear economic and ALARA benefits, but poses some technical and regulatory challenges. BNFL Fuel Solutions (BFS) is exploring the possibility of obtaining U.S. Nuclear Regulatory Commission (NRC) approval to transport the existing loaded MSBs using the NRC-certified FuelSolutions™ TS125 transportation cask.

BFS has examined alternative approaches for licensing the existing loaded MSBs for transportation, including burn-up credit (BUC) and moderator exclusion. Based on the preliminary assessment, in which the design-basis storage payload was considered, BFS concluded that analysis based on BUC that takes credit for actinides only (actinide-only BUC) would not be sufficient to qualify the MSBs in accordance with the transportation regulations. The initial study was supplemented by additional evaluations that model the known characteristics of the SNF assemblies in each MSB (i.e., MSB-specific evaluations), rather than modeling the bounding, design-basis SNF-assembly characteristics.

This paper summarizes the results of the preliminary MSB-specific evaluation using three types of BUC analysis, each with different assumptions: actinide-only BUC, which takes credit for actinides only; limited-fission-product BUC (5-FP), which takes credit for five major fission-product isotopes plus eleven major actinide isotopes; and full-fission-product BUC (full-FP), which takes credit for fifteen fission-product isotopes plus eleven major actinide isotopes. The results of each analysis were measured against k_{eff} limits of 0.95 and 0.98, respectively. A risk-based licensing framework for BUC analyses that are more aggressive than current NRC-guidelines allow is also discussed.

2. Physical Description of VSC-24 and Transportation Cask TS125

The primary components of the VSC-24 storage system include the Ventilated Concrete Cask (VCC) and the MSB (canister). The MSB is stored inside the VCC. The VCC provides biological shielding, features to permit natural convective cooling, and physical protection of the MSB and its SNF assembly payload during storage. The MSB provides confinement of radioactive nuclides, structural support of the SNF assemblies, and criticality control. The structural components of the MSB shell assembly are fabricated entirely from SA-516, Grade 70 carbon steel. These include a 19 mm (¾ in.) thick bottom plate, 25 mm (1 in.) thick x ϕ 159 cm (62.5 in.) (OD) cylindrical shell, shield lid, and 76 mm (3 in.) thick structural lid. The MSB storage sleeve assembly, which is fabricated entirely from SA-516, Grade 70 carbon steel, consists of 24 storage sleeves and three support plates 71 cm (28 in.) long. A cross-sectional view of the MSB storage sleeve assembly through a support plate section is shown in Figure 1. Each storage sleeve assembly is a square tube with an outer dimension of 234 mm (9.2 in.) and a wall thickness of 5 mm (0.2 in.). Each of the 58 VSC-24 MSBs houses one of three different MSB storage sleeve lengths: long (404 cm or 159.0 in.), standard (416 cm or 163.6 in.), or short (375 cm or 147.5 in.).

The FuelSolutions™ TS125 Transportation Package comprises a TS125 cask, impact limiters that are bolted to the top and bottom ends of the TS125 cask, and a canister payload (i.e., a canister loaded with SNF assemblies). The TS125 Transportation Package has an overall length of 870 cm (342.4 in.), an outside diameter of 365 cm

(143.5 in.), and a maximum gross weight of 130 tonnes (285,000 lb.). Payloads weighing up to 38.6 tonnes (85,000 lb.) can be accommodated in the TS125 cask design. The cask body and closure lid are constructed of Type-XM-19 stainless-steel material. Lead gamma shielding is sandwiched between the inner and outer shells. In the shipping mode, the cask is equipped with impact limiters. A schematic of the FuelSolutions™ TS125 transportation cask without the impact limiters is shown in Figure 2.

The FuelSolutions™ TS125 transportation cask is designed and licensed to transport nearly all domestic commercial pressurized water reactors (PWR) SNF assembly classes in a FuelSolutions™ W21 canister and BWR SNF assemblies used at Big Rock Point Nuclear Plant in the United States in a FuelSolutions™ W74 canister. The maximum payload weight (i.e., combined weight of the canister, SNF assemblies, and cask cavity spacer, if required) of the TS125 cask is limited to 38.6 tonnes (85,000 lb.). The design basis decay heat load for the TS125 cask payload is 22 kW.

3. Criticality Evaluation

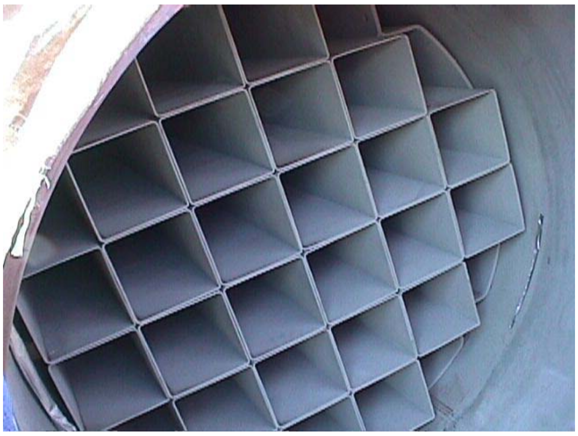

Figure 1. Cross Section of a VSC-24 Multi-Assembly Sealed Basket (MSB)

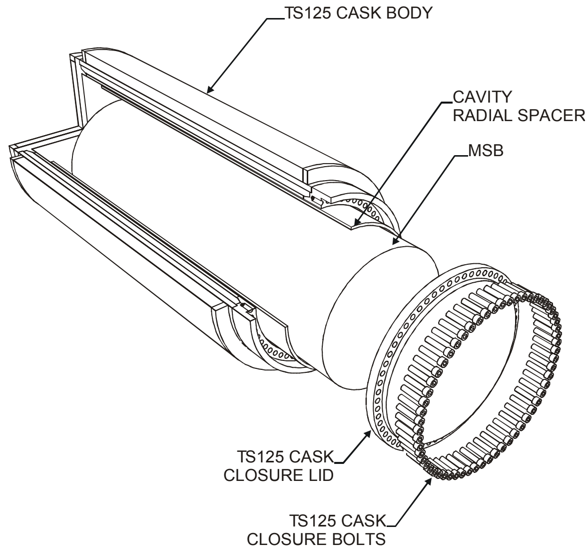

Figure 2. Schematic of a TS125 Transportation cask

Preliminary MSB-specific burnup credit (BUC) criticality analyses were performed for the existing loaded MSBs at the three plant sites (Point Beach, Palisades, and ANO) to evaluate the potential use of the MSB-specific BUC-analysis approach for licensing the existing MSBs for transportation. The preliminary analyses were performed to determine the number of MSBs from each plant that can be qualified for transportation with a range of BUC "levels" and acceptance criteria. The preliminary analyses considered three BUC levels: (1) Actinide-only BUC, (2) 5-FP BUC, and (3) full-FP BUC. For 5-FP BUC, five fission products were selected based on their relative worth. (However, for the final licensing analyses, a set of five fission products would be selected based on the availability of chemical assay data.) The preliminary MSB-specific BUC evaluation also considered MSB qualification based on a reduction of the standard 5% regulatory safety margin to 2% (i.e., k_{eff} < 0.98).

Unlike typical BUC criticality analyses, which determine limiting allowable parameters for any fuel assembly to be loaded into a canister (e.g. minimum allowable burnup vs. initial enrichment), the MSB-specific BUC criticality analyses are performed using the known characteristics of each individual fuel assembly in each MSB. For each fuel assembly, the fuel material composition is determined based upon the known assembly burnup, initial enrichment, and cooling time. The effects of the axial burnup profile are also modeled.

The explicit approach described above removes most of the unnecessary conservatism present in the typical BUC criticality analysis approach. All of the assembly-specific material properties are modeled accurately, not conservatively. Credit is taken for all of the burnup present in every assembly (and the resulting reduction in reactivity), as opposed to only taking credit for the burnup present in the limiting (most reactive) assembly in the basket (as is effectively the case with the standard BUC approach). In other words, full credit is taken for the less reactive assemblies' ability to compensate for the presence of the more reactive assemblies.

Another area where unnecessary conservatism is reduced is the modeled assembly cooling time. Standard BUC analyses conservatively model a cooling time of 5 years. For the MSB-specific BUC analysis, the actual cooling time of each loaded assembly (as of 2012, the MSB shipping date selected as the basis of the preliminary analyses) is explicitly modeled. Since the actual cooling times of the fuel assemblies loaded in the MSBs (as of 2012) range from 20-30 years, this more accurate approach results in a significant reduction in calculated MSB reactivity. Credit is also taken for water displacement by stored BPRA inserts.

3.1 Material Composition Calculations

Isotopic compositions are calculated for the fuel material present in all fuel assemblies in each MSB based on the assembly average burnup, initial enrichment, and discharge date obtained from plant records. Assembly cooling times used for the isotopic calculations are based on shipment in the year 2012.

In order to account for end effects, the preliminary MSB-specific BUC analyses are performed using the three limiting axial burnup profiles from DOE/RW-0472 [1]. The limiting axial burnup profiles are normalized profiles with 18 equal-length axial zones. The axial burnup profile modeled for each fuel assembly is determined based on the limiting axial burnup profile corresponding to the assembly's average burnup level. The local burnup level in each of the 18 axial zones of the assembly is calculated by multiplying the assembly average burnup level by the normalized burnup level of the zone from the corresponding limiting axial burnup profile. The material composition within each axial zone is determined based upon the local burnup level, the assembly's initial enrichment, and the assembly's cooling time.

The spent fuel material compositions for each axial zone of all fuel assemblies in each MSB were calculated using the ORIGEN 2.1 point-depletion code [2]. The "pwrus" ORIGEN 2 weighted cross-section library was used for fuel with local burnup levels under 34 GWd/MTU, and the "pwrue" cross-section library was used for burnup levels over 34 GWd/MTU. These ORIGEN 2 weighted cross-section libraries are tailored to yield accurate results for most PWR fuel assemblies. The point-depletion calculations were conservatively performed assuming a single, continuous irradiation period (which maximizes reactivity of the spent fuel material mixture). A representative assembly thermal power level of approximately 40 MW/assembly was assumed.

ORIGEN 2.1 isotopic mass calculations were performed for 18 axial zones of 24 fuel assemblies in each of 58 loaded MSBs. The ORIGEN 2.1 output files give the masses (in grams/MTU) for all isotopes present in the spent fuel material (not just the isotopes modeled in the criticality analyses). The masses of certain isotopes are then selected and extracted from the ORIGEN 2.1 output, based upon the set of isotopes that are to be modeled in a given criticality analysis.

3.2 Criticality Calculations

The MSB-specific BUC criticality analyses were performed using the MCNP4C Monte Carlo code [3]. An explicit, three-dimensional MCNP analysis was performed on each specific MSB, with its associated spent fuel payload. The criticality analyses model each MSB inside the FuelSolutions™ TS125 transportation cask. Reflective boundaries are placed around the cask to conservatively model an infinite cask array. The criticality analyses model the axial span covered by the fuel zone of the loaded assemblies. Infinite water reflectors are conservatively placed above and below the fuel zone configuration.

The fuel assembly class (i.e., W 14x14, CE 15x15, B&W 15x15, or CE 16x16) stored in each MSB was determined based on plant records. Minor geometry variations that exist within most assembly classes were not considered for the preliminary evaluation. Representative assembly geometries within each fuel class were used because the effects of geometry variations within each assembly class on overall MSB reactivity are expected to be minor. The fuel assembly models include the fuel rods and guide tubes, but not the grid spacers, which are conservatively modeled as water. The fuel rod materials are modeled based on the isotopic compositions determined in the fuel depletion analysis. The density of each isotope is conservatively calculated based on an assumed UO_2 density of 10.52 g/cc (i.e., 96% of UO_2 theoretical). All fuel rod array locations are conservatively modeled containing fuel rods. The analyses model the solid zircaloy guide bars present in Palisades' fuel. In eight of the loaded MSBs at the ANO site, all of the loaded B&W 15x15 assemblies contain inserted BPRA assemblies. The criticality analyses for these eight MSBs model the presence of the BPRA assemblies. The poison material in the BPRA assemblies is modeled as $^{11}B_4C$. Thus the analyses take credit for the water displacement of the BPRA assemblies, but do not take credit for any poison material that may be present. BPRA assemblies are the only control components (i.e., assembly inserts) licensed for storage in the VSC-24 cask. Thus, no other types of assembly insert are present.

A criticality sensitivity analysis was not performed to determine the most reactive configuration for the preliminary MSB-specific BUC evaluation. Instead, the most reactive configuration from the storage criticality evaluation was used for the preliminary MSB-specific BUC evaluation, including the set of most reactive tolerance dimensions and all fuel assemblies positioned towards the center of the basket. The analyses also conservatively assumed fresh water at full density, and water in the pellet/clad gap.

For each MSB, a separate BUC criticality analysis was performed for each of the three assumed BUC levels: actinide-only BUC, 5-FP BUC, and full-FP BUC.

3.3 Estimated Δk_{eff} Values

The purpose of the preliminary MSB-specific BUC evaluations was to provide a rough estimate of the number of loaded MSBs that may qualify under various analysis assumptions and criteria. Accordingly, rigorous benchmarks of the codes used to perform the isotopic and criticality analyses were not included in the scope of the preliminary MSB-specific BUC analysis. Rather, the effects of code bias and uncertainty on the neutron multiplication factor (k_{eff}) were estimated based on published data from similar spent fuel transportation package BUC analyses.

The effects of fuel depletion code bias and uncertainty on the package reactivity, in terms of Δk_{eff}, were estimated based on the results of published data from other fission-product BUC evaluations. The published data includes Δk_{eff} values for full-FP BUC, defined as the difference between the k_{eff} values calculated using nominal (“best-guess”) isotopic concentrations and the k_{eff} values based on “licensing basis” isotopics. From this data, overall Δk_{eff} values were estimated as a function of assembly burnup level, and these values were applied to the preliminary MSB-specific BUC analysis results.

To estimate the Δk_{eff} values for the actinide-only and 5-FP BUC analyses, the full-FP Δk_{eff} values are scaled down based upon the relative overall absorptivity of the set of nuclides in question. This approach is based upon the assumption that if a set of nuclides affects (i.e., reduces) the calculated k_{eff} value by a smaller amount, a smaller Δk_{eff} value is sufficient to account for the effects of uncertainties in the concentrations of those nuclides. The effect on k_{eff} of a given nuclide is proportional to its absorptivity.

Published data shows that for typical PWR fuel assemblies under full-density water moderation, the overall absorptivity of the actinide isotopes is ~58% of the total absorptivity of all of the isotopes modeled in a full-FP BUC evaluation. The overall absorptivity of the actinides along with the “top five” fission products is ~84% of the absorptivity of the full fission product case. Thus, to estimate the Δk_{eff} values that apply for the actinide-only and 5-FP BUC analyses, the burnup-dependent Δk_{eff} values determined for the full-FP BUC case are multiplied by factors of 0.58 and 0.84, respectively.

It is assumed that the magnitude of the actual Δk_{eff} (much like the actual k_{eff} value itself) will scale approximately with the average values of the parameters within the basket. Thus, the Δk_{eff} values for each MSB were estimated based on the average of assembly average burnup level for all 24 fuel assemblies in each MSB (i.e., payload average BU). The resulting estimated Δk_{eff} values are added to the “raw k_{eff}” values from the preliminary criticality analyses to estimate the overall k_{eff} values.

3.4 Preliminary MSB-Specific BUC Evaluation Results

The results of the preliminary MSB-specific BUC evaluation show that the vast majority of the 58 loaded MSBs can be qualified for transport using the MSB-specific BUC approach. The preliminary evaluation results show that, while only a few of the 58 loaded MSBs can be qualified within established NRC guidelines [4] (i.e., actinide-only BUC, k_{eff} < 0.95), approximately two-thirds of the MSBs can be qualified using a slightly more aggressive approach (i.e., actinide-only BUC, k_{eff} < 0.98). Maximum k_{eff} values for all 58 MSBs are presented in Figure 3 for all three

BUC levels evaluated. The results from the 5-FP BUC analyses show that 43 MSBs qualify with k_{eff} values lower than 0.95, and all but 6 MSBs qualify with k_{eff} values lower than 0.98. Finally, the results of the full-FP BUC analyses show that 49 MSBs qualify with k_{eff} values lower than 0.95 and all except one MSB qualify with k_{eff} values lower than 0.98.

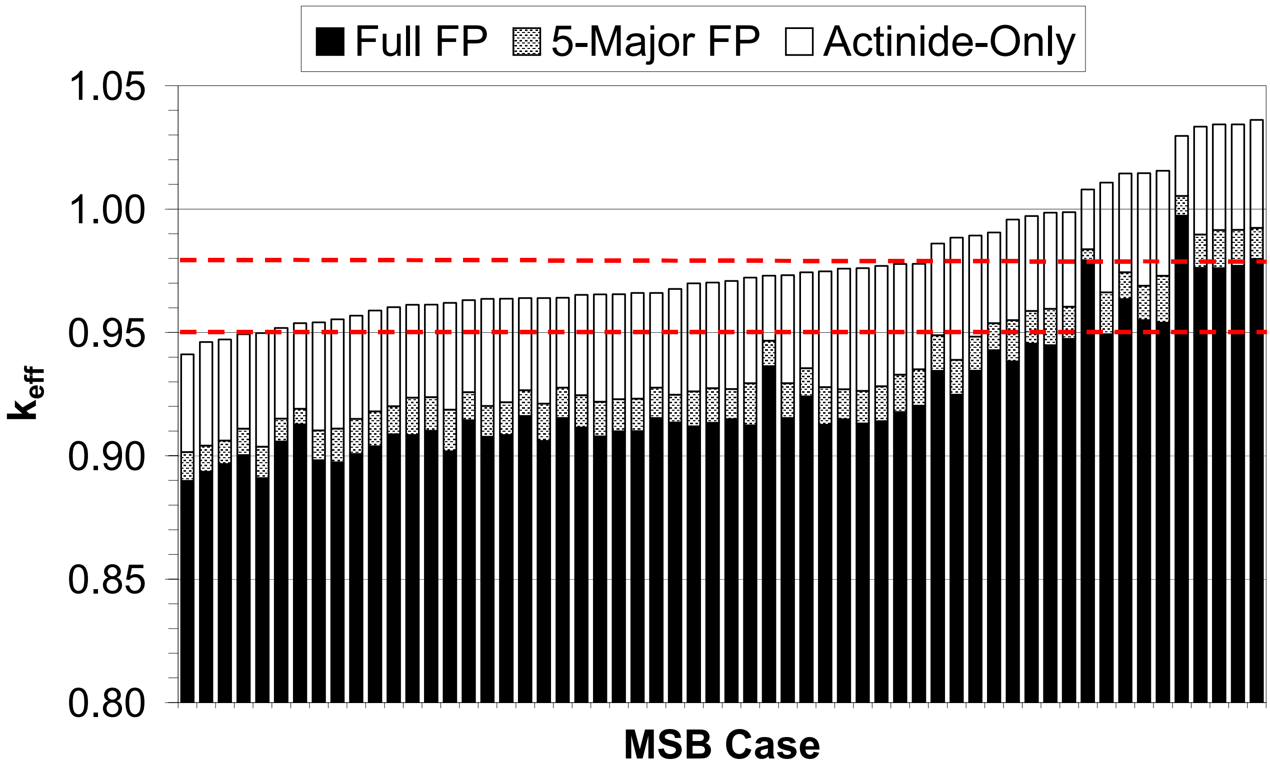

Figure 3. Preliminary MSB-Specific BUC Analysis Results

4. Risk-Based Licensing Approach

4.1 Background

The traditional licensing of transportation cask and canisters in the United States is based on the 10 CFR 71 [5] regulations that are very similar to the IAEA Standard TS-R-1 [6]. The NRC has developed various internal guidance documents (Standard Review Plan, Regulatory Guides and Interim Staff Guidance) for its use in the licensing process. The traditional licensing approach is based on a “deterministic” approach that establishes requirements for engineering margin and quality assurance in design, manufacture, and construction.

In 1995, NRC issued [7] a policy statement on the use of probabilistic risk assessment (PRA) encouraging the greater use of this method to improve safety decision-making and improve regulatory efficiency. Subsequently, Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to The Licensing Basis” was issued [8]. This regulatory guide provides the NRC staff’s recommendations for using risk information in support of licensee-initiated licensing basis changes to a nuclear power plant. Though the regulatory guide applies only to the reactor systems, the general principles and methodology are applicable to spent fuel cask storage and transport systems.

The application of risk assessment methods to the regulation of nuclear material and waste disposal are addressed in SECY 98-138, “Risk-Informed, Performance-Based and Risk-Informed, Less-Prescriptive Regulation in the Office of Nuclear Material Safety and Safeguards” [9], and SECY-99-100, “Framework for Risk-Informed Regulation in the Office of NMSS” [10]. SECY 98-138 points out the practical differences between the application of risk-based approaches to the reactor regulation and the nuclear material side of the NRC. The staff acknowledged that the PRA and other risk-graded approaches would be encouraged for determining high and low-risk activities. SECY 99-100 addresses the commitments made by the staff in SECY 98-138. Regarding application of risk-based approaches in the transport of nuclear fissile material, the staff noted the following activities: revalidation of the “Modal Study” contained in NUREG-0170 [11]; and application of PRA methodology to approve the one-time shipment of the Trojan Reactor Pressure Vessel [12].

The Trojan application requested an exemption from the requirements of 10CFR71.71(c)(1) and 10CFR71.73(c)(1). The applicant, Portland General Electric (PGE), contended that the Trojan Reactor Vessel Package (TRVP) would remain oriented in an horizontal position during and after the normal drop of 0.3m (1 ft) and thus requested exemption from subparagraph 10CFR71.71(c)(1) that requires an evaluation of the drop in any orientation. Additionally, PGE also stated that the maximum credible distance the TRVP could drop during a hypothetical transport accident is only 3.35m (11 ft) rather than the regulatory requirement of 10m (30 ft). PGE had performed a probabilistic safety study to support the exemption request.

4.2 Application of Risk-Based Licensing Approach to VSC-24 Transportation

As discussed previously, transportation certification based on BUC approaches that are more aggressive than those recommended by current NRC guidelines may be possible for existing loaded canisters with well-characterized spent fuel payloads. However, NRC approval of these more aggressive BUC approaches will rely on the ability to demonstrate that the risks posed by the more aggressive BUC analyses and/or criticality margins are still within acceptable limits (i.e., a risk-based approach). Such an approach requires risk analyses to evaluate the possibility and consequences of an accidental criticality event.

A risk-based “waterfall” approach is suggested for acceptance of more-aggressive BUC approaches. The risk-waterfall acceptance approach, which is best illustrated by the risk-waterfall diagram shown in Figure 4, minimizes the overall risk by seeking NRC certification for each MSB based on the lowest-risk qualifying BUC approach.

In the waterfall diagram the various BUC approaches are listed with increasing licensing risk. It is seen that with the use of the actinides alone, five MSBs could be qualified. As stated earlier, this approach is already accepted by the NRC [4] and thus has low risk. Next with the use of five fission products (and actinides), thirty-eight MSBs would meet the acceptance criterion of $k_{eff} < 0.95$. Though, the NRC is yet to approve this approach, published literature [13,14] indicates that there is technical basis for taking credit for the limited fission products. Also, referring back to Figure 3, it is noted that with taking credit for actinides only but using an acceptance criterion of $k_{eff} < 0.98$, thirty-five of the above thirty-eight additional MSBs would be qualified. Use of the higher k_{eff} limit could be justified as no credit is taken for burnup of any of the fission products. Thus, an additional thirty-five to thirty-eight MSBs could be qualified with minimal licensing risk.

Fourteen of the fifteen remaining fifteen MSBs could be qualified with consideration of full fission products, nine with $k_{eff} < 0.95$ and five with $k_{eff} < 0.98$. It is recognized that additional, quantitative risk analyses may be required considering the specific payloads, transportation routes, dedicated rail transport, and other administrative controls. A methodology similar to the one presented in the IAEA proceedings of a technical committee meeting entitled, "Practices and Developments in Spent Fuel Burnup Credit Applications“ [14] may be used to calculate the risk. In these proceedings, two cases were considered: the first addresses the relative probability of exceeding the criticality limit with and without BUC and the second case examines the effect of using BUC on the overall risk for dry fuel transport.

This risk-based waterfall approach also helps put into perspective the balance that must be considered between the risks associated with more-aggressive BUC approaches versus the risks associated with the alternatives, namely unloading canisters and repackaging fuel assemblies for shipment or relying on moderator exclusion to license the canisters for shipment.

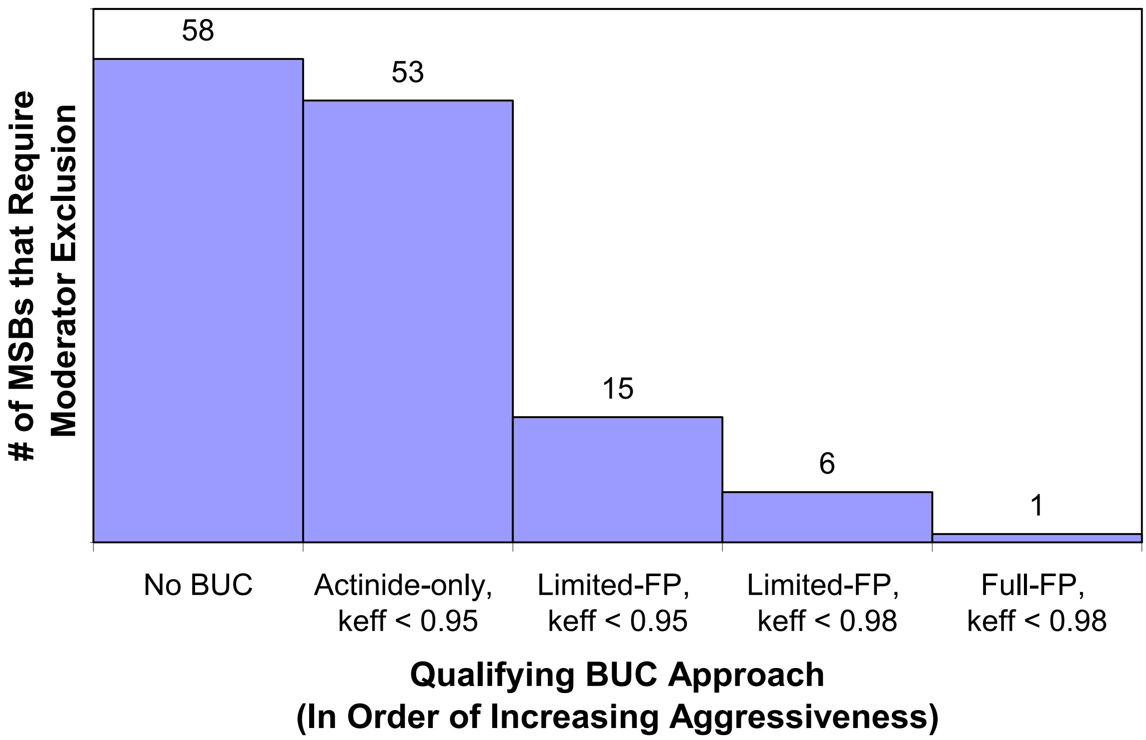

Figure 4. Risk Waterfall Diagram

5. Conclusions

The eventual disposition of the spent fuel assemblies loaded in canisters and casks currently designed and licensed only for on-site storage is an industry-wide issue. The canister-specific BUC evaluation approach developed by BFS can be used to license many of these storage canisters and casks for transportation. This will allow these storage canisters and casks to be transported intact to a long-term storage facility or repository, thereby minimizing fuel handling operations, impact on plant operations, and occupational exposure, as well as total infrastructure costs.

Application of the proposed canister-specific BUC analysis approach to a preliminary evaluation of the 58 loaded MSBs demonstrates the benefits of this approach. The results of this preliminary evaluation show that a more rigorous analysis based on the known characteristics of the loaded spent fuel, rather than the design-basis fuel parameters, produces significantly lower maximum k_{eff} values and can be used to qualify many of the existing loaded storage canisters for transportation.

Transportation certification for storage canisters having more reactive spent fuel payloads may require reliance on BUC approaches that are more aggressive than current NRC guidelines allow. Credit may be required for fission-product isotopes that do not have sufficient chemical assay data for benchmarking. In addition, reduced criticality safety margins may be required. For these more-aggressive BUC approaches, a risk assessment should be provided to support the NRC-approval basis. The risk assessment should evaluate the possibility and consequences of an accidental criticality event based upon inaccuracies in the characterization of the spent-fuel payloads.

6. References

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