NAC INTERNATIONAL, INC.

NAC-UMS® UNIVERSAL STORAGE SYSTEM

PRELIMINARY SAFETY EVALUATION REPORT

AMENDMENT NO. 3

PRELIMINARY SAFETY EVALUATION REPORT

Docket No. 72-1015
NAC-UMS® STORAGE SYSTEM
Certificate of Compliance No. 1015
Amendment No. 3

SUMMARY

By application dated January 15, 2002, as supplemented on February 4, 2002, July 3, 2002, August 7, 2002, November 27, 2002, December 11, 2002, and August 15, 2003, NAC International, Inc. (NAC) requested approval of an amendment, under the provisions of 10 CFR Part 72, Subpart K, to Certificate of Compliance No. 1015 for the NAC-UMS® Universal Storage System to incorporate Enhanced Design Features (E-UMS).

NAC requested changes to the Certificate of Compliance (CoC), including its attachments, and revision of the Final Safety Analysis Report (FSAR). The requested changes incorporated into this amendment were (1) to add an alternate poison material, i.e., METAMIC; (2) to increase the boiling water reactor (BWR) fuel assembly weight (from 696 pounds to 702 pounds); (3) to revise fuel assembly dimensions (length and width) for more comprehensive coverage (BWR and pressurized water reactor (PWR)); (4) to revise thermal analyses to extend operating time limits for vacuum drying, canister in transfer cask, helium backfill, forced air cooling, and in pool cooling; (5) to revise allowable fuel cladding temperature to reflect "Cladding Considerations for the Transport and Storage of Spent Fuel," Interim Staff Guidance -11, Revision 2 (ISG-11); (6) to incorporate criticality analyses for loading 5.0 weight % enriched PWR fuel with a minimum of 1000 ppm soluble boron; (7) to reorganize the Criticality Section to separately describe SCALE and MONK computer programs; (8) to delete annual effluent reporting requirement; and (9) to incorporate editorial and administrative changes.

The Nuclear Regulatory Commission (NRC) staff has reviewed the application using the guidance provided in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems." Based on the statements and representations in the application, as supplemented, the staff concludes that the NAC-UMS® Enhanced Universal Storage System, as amended, meets the requirements of 10 CFR Part 72. The changes to the CoC are indicated by change bars in the margins.

BACKGROUND

The NAC-UMS® system consists of the following components: (1) stainless steel transportable storage canister (TSC), which contains the spent fuel; (2) vertical concrete cask (VCC), which contains the TSC during storage; and (3) a transfer cask, which contains the TSC during loading, unloading and transfer operations. Each TSC stores up to 24 fuel assemblies from a PWR or up to 56 fuel assemblies from a BWR.

The NAC-UMS® was originally approved for storage of site specific spent fuel from the Maine Yankee plant. Amendment No. 2 to the NAC-UMS CoC allowed various Maine Yankee spent fuel components associated with assemblies as authorized contents of the NAC-UMS system. The amendment also revised the CoC format for consistency and revised the CoC Appendix A, Technical Specifications (TS) and Appendix B, Approved Contents and Design Features.

STRUCTURAL EVALUATION

This section provides the staff evaluation of the structural performance of the NAC-UMS® system for enhanced structural design features and design criteria. It also evaluates the thermal stress re-analysis of structural components in conjunction with individual and combined effects of dead weight, pressure, and handling loads.

Structural Design Features and Design Criteria

Advanced Transfer Cask

To increase the lifting capability from a design basis content weight of 88,000 lbs to 98,000 lbs, the Safety Analysis Report (SAR) modifies slightly the approved Standard transfer cask to result in the Advanced transfer cask configuration. The Advanced transfer cask is identical to the Standard transfer cask, except that the Advanced transfer cask incorporates a 0.75-inch thick support plate welded, above each of the trunnions, between the inner shell and the outer shell to achieve a more favorable load-transferring path and, thus, higher lifting capability.

SAR Section 3.4.3.4.1 follows the same modeling approach and includes the structural details of the 0.75-inch thick support plate to perform a finite element analysis of the trunnion-to-shell joint of the Advanced transfer cask. For the purpose of structural analysis, the loaded transfer cask is assumed to weigh 225,000 lbs. This corresponds to an assumed transfer cask content of 103,000 lbs, which envelopes the design basis content of 98,000 lbs. SAR Tables 3.4.3.4-1 and -2 summarize stress results for the top and bottom surfaces of the transfer cask outer shell, respectively. SAR Tables 3.4.3.4-3 and -4 summarize the corresponding results for the inner shell. Except for localized over-stresses, as permitted by ANSI N14.6, “Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More,” stresses in the shells are shown to meet the stress design factors of 6 and 10 against the respective yield and ultimate strengths. The SAR provides calculations for a maximum linearized trunnion bending stress of 4,260 psi, which corresponds to the stress design factors of 7.5 and 16.6 against the respective material yield and ultimate strengths.

Increased BWR Fuel Assembly Weight

SAR Table 12B-1, Fuel Assembly Limits, lists the BWR fuel assembly weight to be equal or less than 702 lbs. SAR Section 11.2.12.4.2 recognizes the upgraded weight, together with fuel tubes and aluminum heat transfer disks, in determining the combined weight for which a 30 g equivalent inertia force is to be applied on the support disk slots. Since the combined weight and inertia load continue to be enveloped by those previously approved, no new structural evaluation needs to be performed to demonstrate structural adequacy of the support disk.

Design Earthquakes

Section B 3.4.1.3 of Appendix B to the CoC lists the upgraded design earthquake levels of 0.26 g and 0.29 g at the top surface of the independent spent fuel storage installation (ISFSI) pad or at the center of gravity of the loaded concrete cask on the ISFSI pad. The design earthquake levels are based on seismic stability analysis of the loaded vertical concrete cask against tipover and sliding. NAC revised the seismic stability analysis, in SAR Section 11.2.8.2, by matching the vertical seismic acceleration component to that of the horizontal. Considering friction coefficients achieved between the steel bottom plate of the concrete cask and concrete surface of the storage pad, two design earthquake levels at 0.26 g and 0.29 g were evaluated. The analysis shows that seismic accelerations for cask sliding are more limiting than tipover. At the same friction coefficient of 0.35 considered in the previous approval, the SAR determines a minimum seismic accelerations of 0.29 g to cause the cask to slide. For a design earthquake level of 0.26 g, it corresponds to a sliding stability margin of 1.12 (0.29/0.26 = 1.12), which satisfies the required margin of 1.1, per ANSI/ANS-57.9, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)." At a higher friction coefficient of 0.40, the SAR calculates that the cask would not slide for a seismic acceleration at less than 0.32 g. This corresponds to a margin of 1.10 (0.32/0.29 =1.10) for a design earthquake of 0.29 g, and is acceptable.

Allowable Temperatures, Aluminum Heat Transfer Disks

SAR Table 4.1-3, Maximum Allowable Materials Temperature, provides material temperature limits for long and short term conditions. The SAR amendment proposed to increase the short term temperature limit for the Aluminum Alloy 6061-T651 heat transfer disks from 700°F to 750°F. The melting range for this alloy is 1080°F - 1206°F. Hot working may be done in the temperature range of 500° to 700°F and hot forged in the temperature range of 750°F to 900°F. The heat transfer disks are not structural components and they are restrained by tie rod spacers and fuel tubes. The aluminum heat transfer disks will withstand the increased short term temperature and continue to perform their function.

Re-analysis of NAC-UMS System Components

SAR Section 3.4.4 uses revised temperature gradients to perform a thermal stress re-analysis. The resulting thermal analyses provides the justification for extending the transfer cask operating time limits by demonstrating structural performance of system components under individual and combined effects of dead weight, thermal, pressure, and handling loads.

Transportable Storage Canister

SAR Table 3.4.4.1-1 summarizes the maximum canister thermal stresses under the normal operating condition. SAR Tables 3.4.4.1-2 and -3 present the respective canister primary membrane and primary membrane-plus-bending stress results for the dead weight load. SAR Tables 3.4.4.1-9 and -10 present stress results for the canister subject to an internal pressure of 15 psig. SAR Tables 3.4.4.1-6, -7 and -8 list stresses under the combined load of normal handling and internal pressure. For the load combinations evaluation, the SAR reports a minimum stress margin of 0.08, which occurs in the canister shell and is acceptable.

Fuel Basket Support Disk

SAR Section 3.4.4.1.8 re-analyzes the PWR and BWR fuel basket support disks for the storage and handling condition. The analysis considers the out-of-plane dead weight and the temperatures at the center and around the outer edge of the fuel support disks. SAR Tables 3.4.4.1-12 and -13 summarize stress results, including thermal stress effects, for the PWR support disk with a minimum stress margin of 2.42. The corresponding results, with a minimum stress margin of 3.58, are reported in SAR Tables 3.4.4.1-15 and -16 for BWR support disks. These stress margins are acceptable.

Fuel Basket Top and Bottom Weldments

SAR Section 3.4.4.1.9 re-analyzes the top and bottom weldments of the PWR and BWR fuel baskets for the storage and handling conditions. The analysis considers appropriate dead weight, support condition, and temperatures at the center and circumference of a fuel basket weldment. For the top weldment of the PWR fuel basket, the temperature at its center is assumed to be 600°F and its circumference 275°F. For the bottom weldment, the corresponding temperatures are 325°F and 175°F. For the top weldment of the BWR fuel basket, the temperature at its center is assumed to be 525°F and at its circumference 225°F. For the bottom weldment, the corresponding temperatures are 475°F and 200°F. SAR Table 3.4.4.1-17 summarizes the load combination effects with a minimum stress margin of 0.07 for the PWR weldments and 0.64 for the BWR weldments. The staff found these results acceptable.

On the basis of the above evaluation, the staff concludes that the proposed changes will not affect the ability of the cask system to meet the requirements of 10 CFR Part 72.

THERMAL EVALUATION

This section provides the thermal evaluation of the NAC-UMS® system for enhanced structural design features and design criteria. The purpose of the evaluation is to ensure that the cask and fuel material temperatures of the NAC-UMS® system will remain within the allowable values or criteria for normal, off-normal, and accident conditions. The evaluation includes confirmation that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation which could lead to gross rupture. The evaluation also confirms that the thermal design of the cask has been analyzed using acceptable analytical and/or testing methods.

The NAC amendment requested that certain time limits associated with loading operations, vacuum drying and transfer to storage pad, be increased to permit more time for conducting these operations before entering the applicable technical specification required actions. The amendment also reduces the required cooling times associated with fuel assembly type, burnup and enrichment, as a result of the new staff guidance provided in Interim Staff Guidance 11, Revision 2 (ISG-11), “Cladding Considerations for the Transportation and Storage of Spent Fuel.” The amendment modifies the effective thermal conductivities of the fuel assemblies and fuel tubes to account for the effects of temperature and media (i.e., water, vacuum, helium or steam) located in the gaps and voids. There is no requested change to the maximum heat loading of the NAC-UMS of 23 kW and as a consequence the maximum temperatures of the various components do not significantly change except for the loading operations of vacuum drying and transfer to storage pad. The applicant has analyzed the NAC-UMS to increase the aforementioned operational time limits for heat loads of 23 kW, 20 kW, 17.6 kW (PWR only), 17.0 (BWR only),14 kW, 11 kW and 8 kW for both PWR and BWR fuels. However, increasing these transfer operational time limits results in the increase of the component temperatures closer to their limits especially for the fuel cladding and heat transfer disk. A summary of the associated component temperatures are provided below for the fuel cladding and the aluminum heat transfer disk, as well as the temperature change from Amendment No. 2.

NAC-UMS PWR (1) Component ➜		Fuel Cladding	Heat Transfer Disk with low burn-up fuel
Allowable Temperature ➜		752°F (for normal, loading & transfer)	750°F (short term) 650°F (long term)
Maximum Temperature During Vacuum Drying or Transfer Operations within the time limits established by the technical specifications (refer to Table 4.4.3-5 of the SAR)	23 kW	724°F (+38°F)	680°F (-6°F)
	20 kW	728°F (+21°F)	664°F (+13°F)
	17.6 kW	731°F (-30°F)	651°F (-23°F)
	14 kW	732°F (-44°F)	630°F (-49°F)
	11 kW	730°F (-62°F)	611°F (-62°F)
	8 kW	731°F (-27°F)	622°F (-27°F)
Maximum Temperature During Vacuum Drying or Transfer Operations, post in-pool or post forced air cooling within the time limits established by the technical specifications (refer to Tables 4.4.3-6 & -7 of the SAR)	23 kW	724°F	680°F
	20 kW	728°F (+14°F)	664°F (+14°F)
	17.6 kW	731°F (+31°F)	651°F (+14°F)
	14 kW	732°F (+1°F)	630°F (+8°F)
	11 kW	730°F (+24°F)	611°F (+34°F)
	8 kW	731°F (+56°F)	595°F (+86°F)

(1) BWR fuel is not listed since it is bounded by PWR
(2) Numbers in ( ) indicate temperature change from Amendment #2.

The staff issued ISG-11, Revision 2, during the review of this amendment. The ISG changed the cladding temperature limit to 400°C (752°F) for normal conditions, which is less restrictive than the previously issued guidance contained in PNL-6189, “Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircalloy Clad Fuel Rods in Inert Gas,” which limited temperatures between approximately 330°C and 380°C depending principally on cooling time and burn-up. ISG-11, Revision 2 also changed the cladding temperature limit to 400°C for short term operations including cask drying; however, the criteria was subsequently amended via a staff analysis for low burnup fuel (i.e., < 45 Gwd/MTU). The results of the staff analysis allows loading of spent fuel with burn-ups less than 45 Gwd/MTU and hoop stresses less than 90 MPa at temperatures up to 570°C (1058°F), the previously accepted short term temperature limit. NAC has not requested this option as a part of this amendment, but instead has utilized 400°C as the temperature limit for the cladding under normal conditions. For high burn-up fuel (> 45 Gwd/MTU), the cladding temperature limit was established at 400°C and thermal cycling limited to less than 65°C for less than 10 cycles at normal conditions. The cladding temperature limit for off-normal and accident conditions is 570°C.

Based on the ISG guidance, the applicant has demonstrated that the cladding remains below 400°C for normal conditions including all loading and transfer operations. For the Maine Yankee high burnup fuel the 400°C limit was also maintained, but no specific discussion was provided regarding the limiting of thermal cycling during loading and transfer operations. However, the staff does not believe it is likely that the thermal cycling limitations could be exceeded during loading and transfer operations. The staff reviewed the aforementioned changes and found them to be acceptable.

The staff has reviewed and confirmed by analysis the decay heat loads and associated cooling times for selected fuel assemblies identified on SAR Tables 2.1.1-2, 2.1.2-2, and 2.1.3.1-4 for PWR, BWR and Maine Yankee site specific fuel assemblies, respectively. Additionally, the staff reviewed the applicant's proprietary calculation for the effective thermal conductivity of the PWR fuel assembly and fuel tube. Different effective thermal conductivities were determined for the varying types of fluids occupying the gaps and voids. The staff also reviewed the applicant's proprietary calculation governing the time restrictions for PWR fuel during loading and transfer operations. The staff found that these calculations adequately supported the conclusions presented in the SAR.

Based on the information presented in the SAR, the applicant's analysis, and the staff's independent calculations, the staff finds that this amendment presents reasonable assurance that the storage cask's material temperatures will be maintained below their limits.

CRITICALITY SAFETY

The section provides the criticality evaluation of the NAC-UMS® system design. The purpose of the evaluation is to verify that the amended spent fuel contents remain subcritical under the normal, off-normal, and accident conditions of handling, packaging, transfer, and storage. The applicable regulatory requirements are those in 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c), and 72.236(g).

The staff reviewed the information provided in the amended NAC-UMS SAR to determine whether the NAC-UMS® system continues to fulfill the acceptance criteria listed in Section 6 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems:"

1. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95% confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.

3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.

4. Criticality safety of the cask system should not rely on use of the following credits:

   a. burnup of the fuel*
   b. fuel-related burnable neutron absorbers
   c. more than 75% for fixed neutron absorbers when subject to standard acceptance tests.

   *Note: Since publication of the NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” (SRP), the NRC has developed Interim Staff Guidance 8 (ISG-8), “Limited Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks.” Future revisions of the SRP will incorporate or reference the current staff guidance in this area.

Criticality Design Criteria and Features

The design criterion for criticality safety of the cask system is that the calculated value of the effective neutron multiplication factor, k_{eff}, including biases and uncertainties, shall not exceed 0.95 under normal, off-normal, and accident conditions.

Criticality safety of the amended NAC-UMS® system continues to depend on the geometry of the fuel baskets and the use of fixed neutron absorber panels for absorbing neutrons. The geometry of the basket and the ^{10}B loading of the neutron absorber panels have not significantly changed in the amended NAC-UMS design.

For the loading of PWR fuel, criticality safety may also depend on the presence of soluble boron in the spent fuel pool water. The criticality analysis for PWR spent fuel has been revised to include higher initial enrichment limits for fuel loaded with a minimum soluble boron concentration of 1000 ppm in the spent fuel pool water.

The criticality safety analysis was also revised to include a determination of assembly-specific enrichment limits for both PWR and BWR assemblies. Previous enrichment limits were based on the most reactive PWR and BWR assemblies, the Westinghouse 17x17 OFA and Exxon/ANF/Siemens Power Corp. 9x9 with 79 fuel rods, respectively. The revised analysis determines the maximum enrichment for 14 PWR and 13 BWR bounding assembly groups for which maximum k_{eff} will remain below the calculated upper subcritical limit (USL).

Other NAC-UMS design changes addressed in the criticality analysis include miscellaneous non-fuel hardware in the guide tubes of Maine Yankee site-specific fuel.

Fuel Specification

The fuel assembly characteristics listed in Table 6.2-1 for PWR assemblies and in Table 6.2-2 for BWR assemblies have been revised to include an identifier linking each assembly type to a table in Section 6.4 giving bounding assembly-specific maximum initial enrichments. Tables 6.4-25 and 6.4-26 list assembly-specific maximum initial enrichments for PWR and BWR assemblies, respectively, and Table 6.4-29 lists assembly-specific maximum initial enrichments for PWR assemblies with soluble boron credit. PWR enrichments range from 4.3 to 5.0 wt% ^{235}U, and BWR enrichments range from 4.4 to 4.7 wt% ^{235}U. For PWR assemblies where credit for 1000 ppm dissolved boron is taken, the initial enrichment limit is 5.0 wt% ^{235}U.

The Maine Yankee site-specific fuel specification was revised to include assemblies with non-fuel components inserted in the guide tubes. Section 6.6.1.4 evaluates the reactivity effect of having assemblies with Pu-Be or Sb-Be start-up sources, control element assembly (CEA) fingertips, or 24-inch incore instrumentation (ICI) segments in the guide tubes, loaded in the four corner positions of the NAC-UMS basket.

The staff reviewed the revised fuel parameters considered in the criticality analyses and verified that they are consistent with or bound the parameters in the CoC's fuel specifications and in Sections 1, 2, and 12 of the SAR. All fuel assembly parameters important to criticality safety have been included in the fuel specifications.

Model Specification

<u>Configuration</u>

The assembly-specific maximum initial enrichments are determined by creating an explicit, finite model of the NAC-UMS cask for each PWR and BWR assembly type, using the most reactive mechanical perturbations, geometric tolerances, and fuel assembly lattice dimension variations determined in the previously approved analysis. Each bounding fuel assembly group is placed in this model and evaluated at enrichments ranging from 4.2 to 5.0 wt% ^{235}U, in order to determine the maximum enrichment for which $k_{eff} + 2\sigma$ will be less than the calculated USL. A second model is used to determine the most reactive geometry and moderator density for the soluble boron credit evaluation. Each PWR assembly group is then placed in this most reactive configuration to determine the maximum allowable initial enrichment with soluble boron in the moderator.

The Maine Yankee site-specific fuel model is revised to consider four fuel assemblies with Pu-Be or Sb-Be start-up sources placed in their center guide tubes, loaded only in the four corner assembly positions in the UMS cask. CEA fingertips and 24-inch ICI segments, which may also be inserted in fuel assemblies loaded in corner positions, are conservatively ignored since they contain no fissile or moderating material. The Sb-Be sources are conservatively modeled as 50% water and 50% Be filling the entire center guide tube in each assembly. The Pu-Be sources are modeled as a mixture of 0.77% volume fraction ^{239}Pu, Be, and water, filling the entire center guide tube in each assembly.

The staff reviewed the applicant's models and agrees that they are consistent with the design descriptions in SAR Sections 1 and 2, including the license drawings. Based on the information presented, the staff agrees that the calculations incorporate the most reactive combination of package parameters and dimensional tolerances.

Material Properties

The compositions and densities for the materials used in the criticality safety analysis computer models are provided in Sections 6.3.4 of the amended NAC-UMS SAR. The specification for neutron absorber plates has been revised to allow the use of METAMIC, or other aluminum/boron carbide neutron absorbing material similar to Boral, provided that it meets the previously approved ^{10}B areal density requirements. The applicant's models considered 75% of the specified ^{10}B areal density, in order to bound the effects of neutron channeling between B_4C particles in the neutron absorber plates. Section 9.1.6 of the SAR gives the acceptance tests for the neutron absorber plates.

The staff reviewed the composition and number densities presented in the SAR and found them to be reasonable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

Criticality Analysis

Computer Programs

The applicant's computer analyses for the previously approved configuration of the NAC-UMS® system were performed using the CSAS25 sequence of SCALE4.3, along with the SCALE system's 27-group cross-section library. The principal code used for the revised analyses is the MONK 8A Monte Carlo Program for Nuclear Criticality Safety Analysis with the JEF 2.2-based point energy neutron libraries. This code was used in the finite cask modeling for the assembly-specific enrichment limits, maximum reactivity cask geometry with soluble boron credit, and PWR assembly specific enrichment limits with soluble boron credit.

The staff agrees that the code and cross-section set used by the applicant are appropriate for this particular application and fuel system. The staff performed its independent criticality analyses using the CSAS25 sequence of SCALE4.4a, along with the SCALE system's 44-group cross-section library.

Multiplication Factor

The applicant's criticality analyses show that the k_{eff} in the NAC-UMS will not exceed 0.95 for all fuel loadings and conditions. Results of the applicant's MONK 8A criticality calculations for the bounding assembly groups are given in SAR Section 6.1 and in the tables of Section 6.4. Tables 6.4-25 and 6.4-26 specifically show the maximum allowable initial enrichments and resulting $k_{eff} + 2\sigma$, for each of the bounding PWR and BWR assembly groups, respectively. The criticality analysis for soluble boron credit included an evaluation which considered varying water/H_3BO_3 density and flooded versus dry fuel-clad gap. This evaluation was performed using the most reactive PWR assembly, and found that a fully flooded canister and fuel-clad gap was the most reactive. All PWR assembly groups were modeled as fully flooded with water at a soluble boron loading of 1000 ppm. Table 6.4-29 shows the maximum allowable initial enrichments and resulting $k_{eff} + 2\sigma$ for bounding PWR assembly groups taking credit for soluble boron in the moderator. All resulting values for $k_{eff} + 2\sigma$ are shown to be less than the MONK 8A calculated USL of 0.9426.

For the applicant's consideration of startup sources loaded in Maine Yankee site-specific fuel assemblies loaded in corner positions, considerably more fissile and/or moderator material is modeled than will actually be in the assembly, as discussed in Section 6.6.1.4.1 of the SAR. The results of analyses modeling Sb-Be and Pu-Be sources show that neither has a significant impact on the reactivity of the system.

The staff performed independent SCALE 4.4a criticality calculations for fully loaded packages of PWR and BWR fuels under full and partial flooding conditions, with fresh and borated water. The staff approximated selected PWR and BWR bounding assembly groups in the NAC-UMS cask by modeling an axially reflected slice of the cask containing heat transfer disk, spacer disk, and interspersed water regions. This model created an infinite height cask, effectively eliminating axial leakage of neutrons. The staff modeled mechanical perturbations, geometric tolerances, and fuel assembly lattice dimension variations consistent with the applicant's analysis, and considered varying moderator densities. The staff also performed several analyses considering the addition of 1000 ppm soluble boron in the cask moderator. Results of the staff's calculations were in close agreement with the applicant's k_{eff} results for the selected assembly groups.

Based on the applicant's criticality evaluation, as confirmed by the staff's calculations, the staff concludes that the NAC-UMS will remain subcritical, with an adequate safety margin, under all normal, off-normal, and accident conditions.

Benchmark Comparisons

The applicant performed MONK 8A benchmark calculations on 127 selected critical experiments, chosen, as much as possible, to bound the range of parameters in the NAC-UMS design. The most statistically significant parameters are the soluble boron loading in the moderator, and cluster (assembly) gap thickness. Parameters such as enrichment, fuel pellet diameter, fuel rod pitch, clad diameter, and H/U atomic ratio were also considered in selecting the critical experiments.

Results of the benchmark calculations show no significant trends in the bias. The benchmark analysis yielded a USL of 0.9426. The applicant stated that the benchmark and cask calculations were performed with the same computer codes, cross-section data, and computer hardware. The benchmark models, however, were created by the code developer, potentially introducing bias uncertainty due to individual modeling technique and code input option differences between the benchmark models and the UMS models developed by NAC analysts. The NAC request for additional information (RAI) response provided an analysis showing that all the primary modeling and input options used in the NAC-UMS analysis were represented in the code developer's benchmark models. The most significant code feature affecting evaluation results, choice of neutron cross-section library, was the same (JEF 2.2) in all of the benchmark and UMS models. The few modeling and input options that are not represented in the benchmark models are not expected to significantly affect the uncertainty in the calculation bias.

The staff reviewed the applicant's benchmark analysis and agrees that the critical experiments chosen are relevant to the cask design. The staff found the applicant's method for determining and using the calculation bias to be acceptable and conservative.

All supportive information has been provided in the SAR, primarily in Sections 1, 2, 6, and 12.

Evaluation Findings

Based on the information provided in the SAR and verified by the staff's own confirmatory analyses, the staff concludes that the NAC-UMS® system meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds that: structures, systems, and components (SSCs) important to criticality safety are described in sufficient detail in Sections 1, 2, and 6 of the SAR to enable an evaluation of their effectiveness. The NAC-UMS® system is designed to be subcritical under all credible conditions. The criticality design is based on favorable geometry and fixed neutron poisons. An appraisal of the fixed neutron poisons has shown that they will remain effective for the 20-year storage period. In addition, there is no credible way to lose the fixed neutron poisons; therefore, there is no need to provide a positive means to verify their continued efficacy during the storage period. The analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the storage of spent fuel for 20 years with an adequate margin of safety.

The staff concludes that the criticality design features for the NAC-UMS® system are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the NAC-UMS® system will allow safe storage of spent fuel. In reaching this conclusion, the staff has considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

ACCIDENT ANALYSIS

NAC requested in the amendment application that the removable surface contamination limits on the exterior surface of a loaded canister be increased up to 10,000 dpm/100 cm^2 from beta and gamma sources (β -γ) and 100 dpm/100 cm^2 from alpha sources (α). The staff accepts maintaining removable surface contamination limits below 1000 dpm/100 cm^2 β-y and 20 dpm/100 cm^2 α , which are consistent with Regulatory Guide (RG) 1.86, "Termination of Operating Licenses for Nuclear Reactors." The values of RG 1.86 represent levels that can be achieved with reasonable decontamination methods and are consistent with current radiological control practices for preventing the spread of contamination to clean or uncontrolled areas. However, NAC has demonstrated in their analysis that higher removable surface contamination limits result in minimal impact on off-site doses to the public.

NAC has requested not to decontaminate the canister after loading operations instead of proposing any changes to canister loading operations. Canisters are not expected to have removable surface contamination levels above 10,000 dpm/100 cm^2 β-y and I00 dpm/100 cm^2 α. The removal of the decontamination steps will result in a considerable occupational dose savings to workers, consistent with ALARA practices.

The staff has reviewed the analyses provided by NAC and has reasonable assurance that increasing the removal surface contamination limits to 10,000 dpm/100 cm^2 β-y and 100 dpm/100 cm^2 α has minimal impact on off-site doses and results in a dose savings to workers. The staff also notes that analysis provided in the SAR is generic and these removable surface contamination limits should be evaluated in accordance with the requirements of 10 CFR 72.212.

CONCLUSION

The NRC staff has reviewed the amendment to the SAR for the NAC-UMS® system. Only the affected SRP sections were included in the SER. The remaining sections were not addressed in the amendment application and were not affected. The Certificate of Compliance has been revised to include the NAC requested changes. Those changes include: (1) adding an alternate poison material, i.e., METAMIC; (2) increasing the BWR fuel assembly weight; (3) revising fuel assembly dimensions (length and width); (4) revising thermal analyses to extend operating time limits for transfer operations; (5) revising allowable fuel cladding temperature to reflect ISG-11, Rev. 2 guidance; (6) incorporating criticality analyses for loading enriched PWR fuel; (7) describing SCALE and MONK computer programs separately; (8) deleting the annual effluent reporting requirement; and incorporating editorial and administrative changes. Based on the statements and representations contained in the application, as supplemented, the staff concludes that these changes do not affect ability of the NAC-UMS® Storage System to meet the requirements of 10 CFR Part 72.

Issued with Certificate of Compliance No. 1015, Amendment No. 3 on **DRAFT**.