PRELIMINARY SAFETY EVALUATION REPORT

DOCKET NO. 72-1015

NAC INTERNATIONAL

NAC-UMS UNIVERSAL STORAGE SYSTEM

CERTIFICATE OF COMPLIANCE NO. 1015

AMENDMENT NO. 5

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I. SUMMARY

By application dated September 22, 2006, as supplemented on May 8, September 6, September 10, September 26, and November 30, 2007, and April 23 and May 8, 2008, NAC International, Inc. (NAC) requested U.S. Nuclear Regulatory Commission (NRC) approval of an amendment to Certificate of Compliance (CoC) No. 1015 for the NAC-UMS® Universal Storage System, in accordance with the provisions of 10 CFR Part 72, Subparts K and L. The proposed amendment would revise the CoC for the NAC-UMS® Universal Storage System to incorporate certain high burnup PWR fuel as approved contents and to reflect those changes in the associated Appendix A, Technical Specifications and Appendix B, Approved Contents and Design Features (collectively, the "TS"). In addition, the proposed amendment to the CoC would include several other changes to the TS and the Final Safety Analysis Report (FSAR) to enhance the loading and storage operation of the NAC-UMS® system.

NAC requested changes to the CoC and TS for the NAC-UMS® system, and revision of the FSAR, to allow for storage of high burnup PWR fuel (up to 60 GWd/MTU assembly average burnup). The other proposed amendment changes were: 1) Elimination of the requirement for the use of a tamper-indicating seal on the concrete cask lid bolts; 2) Elimination of the Technical Specification requirement (Limiting Condition for Operation [LCO] A 3.1.5) for the helium leakage test of the canister shield lid to canister shell weld; 3) Revision of the Technical Specification Section B2.1 written reporting requirement to 60 days (was 30 days); 4) Elimination of the requirement for Charpy V-notch impact testing of the 0.625-inch nominal thickness BWR support disk material (SA 533, Type B, Class 2); and 5) Revised Delta note 8 on license drawing 790-585 to make the use of the structural lid and shield lid threaded plugs and dowel pins optional. In its September 6, 2007, response to the NRC staff's request for additional information (RAI) dated May 30, 2007, NAC supplemented its application to: 1) reinstate the helium leak testing requirements; 2) revise the definitions of intact and damaged fuel to conform with the NRC guidance of Interim Staff Guidance ISG-1, Revision 2; and 3) revise the TS requirements to limit potential thermal cycling of the fuel cladding to conform with the NRC staff's position in ISG-11, Revision 3.

The NRC staff has reviewed the amendment application and the proposed supporting changes to the FSAR for the NAC-UMS® Universal Storage System. The staff used the guidance provided in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," (SRP), dated January 1997, and the applicable Interim Staff Guidance (ISG) documents developed by the Spent Fuel Project Office (currently the Division of Spent Fuel Storage and Transportation) in the NRC's Office of Nuclear Material Safety and Safeguards. The ISG documents can be viewed at http://www.nrc.gov/reading-rm/doc-collections/isg/spent-fuel.html. The staff performed a detailed evaluation of the proposed changes, which is documented in this safety evaluation report (SER). This SER addresses only those specific changes requested by NAC in this amendment application; the staff did not perform a review of those design features or analyses that were previously reviewed and approved and are unchanged for this amendment application.

Based on the statements and representations in the application, as supplemented, the staff concludes that the NAC-UMS® Universal Storage System, as amended, meets the applicable requirements of 10 CFR Part 72 for the safe storage of spent fuel, and those of 10 CFR Part 20 for radiation protection. The changes to the CoC and the Technical Specifications incorporated as part of this amendment are indicated by change bars in the margins.

II. STRUCTURAL EVALUATION

The objectives of the structural review are to assess the applicant's safety analysis of the structural design features, the structural design criteria, and the structural analysis methodology used to evaluate the expected structural performance capabilities under normal operations, off-normal operations, accident conditions and natural phenomena events for those structures, systems, and components important to safety included in this amendment.

The structural review was conducted in accordance with the regulations set forth in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified as required by 10 CFR 72.236(a) so that the design basis and the design criteria that must be provided for the structures, systems, and components important to safety can be assessed under the requirements of 10 CFR 72.236(b).

The staff reviewed the amendment request, including proposed FSAR Chapter 11, Section 11.2.16, entitled "Fuel Rods Structural Evaluation for Burnup to 60,000 MWd/MTU," which presented new structural analyses of the fuel rod cladding integrity during end drop and side drop accidents, using material properties of high burnup fuel assemblies. The staff reviewed these revised analyses to assess compliance with 10 CFR Part 72 requirements.

For the side drop accident analyses, NAC used the effective cross-sectional properties of the fuel cladding and the fuel pellet to compute the dynamic load factor (based on the lowest frequency of the extensional mode shape). The staff noted the conservative assumptions used in NAC's previous analyses, but disagreed with the applicant's using the pellet cross-sectional properties to demonstrate integrity of the cladding, as this was not substantiated with any available data for high burnup fuel (RAI 11-4). The applicant revised these calculations using the cross-sectional properties of cladding alone to demonstrate the integrity of the cladding under the end drop and side drop accidents.

For the end drop analysis, NAC performed a transient analysis using LS-DYNA for the PWR fuel rods with a bounding value for the bow of the fuel assembly loaded into the canister (NAC proprietary Calculation No. 71160-2026, Revision 0, submitted November 30, 2007; non-proprietary version submitted May 8, 2008). The end drop acceleration was developed assuming a 24-inch end drop accident condition for the UMS storage cask. The effect of the high burnup condition (60 GWd/MTU) of the fuel rods was taken into account by incorporating an oxide layer 120 microns thick for the PWR clad. The NRC staff evaluated the impact of these assumptions on the analyses and concurs with the conclusion drawn by NAC. The analyses considered the bounding conditions for the NAC-UMS® system.

NAC's revised structural evaluation for the end drop of an intact fuel assembly is based on this analysis. An ANSYS model and solution were developed to provide the coordinates of the fuel clad as input for the LS-DYNA model. This was accomplished by obtaining a static solution with the ANSYS model, and then using the option to update the coordinates of the nodes based on the displacements from the solution. A static force was applied to the ANSYS model at the grid nearest the axial center to develop a 0.55 inch lateral displacement. The cladding was modeled with shell elements. Each grid was modeled using brick elements to maintain the spacing between the fuel rods at the grid. The fuel pellet was not modeled explicitly, but its mass was taken into account in the clad density. The fuel tube was modeled using brick elements to restrict the lateral motion of the fuel assembly. Each of the fuel rods in the ANSYS model was simply supported at each end. Five (5) LS-DYNA models were considered for the 24-inch cask end drop conditions. All models incorporated a bow of 0.55 inch. The PWR fuel assembly evaluation considered the condition of missing grids at the bottom end of the fuel assembly such that the distance to the first grid from the end of the fuel rod was 60 inches. These cases enveloped the range of the cross-sectional moments for the PWR fuel rods and the grid spacing at the bottom of the fuel assembly. A conservative value of 10.89×10^6 psi was used for the modulus of elasticity of the irradiated zirconium fuel pellet, based on 90% of modulus at 150º C. The staff reviewed the ANSYS models and the LS-DYNA analyses and finds the overall analytical approach acceptable for this 10 CFR Part 72 storage application.

For the side drop analysis, NAC used the bounding fuel length rod of 60 inches to envelope all fuel types, and included the case of a missing support grid in the fuel assembly. The total stress in the cladding (including the bending stress, hoop stress and the bearing stress between two fuel rods) was 57.0 ksi for the top fuel rod. The total stress was shown to be less than both the allowable ultimate and the yield stress. However, the referenced calculation did not show that the configuration was evaluated for a 60g side drop load. In response to the staff's RAI (11-6), the applicant furnished the analysis for the side drop load scenario (NAC proprietary Calculation 71160-2025 Revision 1, Appendix C, submitted September 10, 2007; non-proprietary version submitted May 8, 2008).

NAC performed a static analysis using ANSYS with a lateral load of 60g's, to evaluate the response to the side drop for the PWR fuel rods with a missing grid, for which the maximum spacing between supports was 60 inches. The fuel rod was modeled with beam elements having the properties of the fuel cladding, taking into account the reduction of the outer radius by 0.0047 inch (120 microns). NAC calculated the maximum stress for a Westinghouse 15 x 15 fuel assembly to be 48.1 ksi. Using the cladding yield strength of 69.6 ksi (at 752º F) as the allowable stress, NAC determined a factor of safety of 1.45. The staff has reviewed this analysis and concurs with the overall approach and the conclusions presented therein.

The staff reviewed NAC's responses to the RAIs in this area (RAIs 3-1, 11-1 through 11-3, 11-5, and 11-7) and determined that the responses were acceptable and the issues resolved.

The NRC staff has determined that the structural integrity of the fuel cladding for high burnup PWR fuel for normal conditions, and for off-normal and accident events, has been adequately demonstrated. On this basis, the staff concludes that the requested changes to the CoC and FSAR for the NAC-UMS® storage system will not adversely affect the ability of the system to meet the regulatory requirements of 10 CFR Part 72.

III. THERMAL EVALUATION

The objective of the thermal review is to ensure that the NAC-UMS® storage system components and fuel material temperatures will remain within the allowable values for normal, off-normal and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding will be within acceptable limits throughout the transfer and storage periods to protect the cladding against degradation, which could lead to gross rupture, during normal, off-normal, and accident conditions.

NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” Section 4.0, “Thermal Evaluation,” and Interim Staff Guidance document ISG-11, Revision 3, “Cladding Considerations for the Transportation and Storage of Spent Fuel,” November 17, 2003, specify the review criteria to be used by NRC staff in performing technical evaluations of applications under 10 CFR Part 72. These review criteria include confirmation that the thermal design of the cask has been evaluated using acceptable analytical methods.

The applicant requested to add high-burnup PWR fuel up to 60 GWd/MTU (assembly average burnup) as allowable contents to the NAC-UMS® system. The applicant has not requested any design changes to the previously-approved thermal design, nor any change to the maximum assembly decay heats previously approved by NRC for the NAC-UMS. Section 4.4.3 (Maximum Temperatures for PWR and BWR fuel) of the updated FSAR for the NAC-UMS® system demonstrates that maximum temperatures for fuel cladding are within acceptable temperature limits, including thermal cycling during loading operations. The NRC staff ran confirmatory cases using ORIGEN-ARP to verify that PWR assembly decay heat is limited to 0.958 kW. Cases were run while varying the final enrichments between 3.3 and 3.7 wt%. The cooling times were taken from the minimum cooling times given per assembly for an average assembly burnup less than 60 GWd/MTU (Table B2-4 of Appendix B to NAC-UMS® Certificate of Compliance). The total decay heat for individual assemblies was determined to be less than 0.958 kW. As stated above, the limits on burnup and cooling time for the new high-burnup contents do not exceed the individual fuel assembly heat load of 0.958 kw for the PWR fuel type that has been previously approved by NRC. In addition, the total canister heat load is 23 kW.

The staff did not review the previous analysis methodology except for verifying that the fuel temperatures meet the acceptance criteria for fuel cladding. To preclude fuel degradation, the applicant limited the maximum cladding temperature under normal conditions of storage and short-term canister transfer operations to 752°F (400°C), which is consistent with ISG-11, Revision 3. The maximum cladding temperature for off-normal and accident events is limited to 1058°F (570°C). Thermal cycles during system drying operations that exceed 117°F (65°C) will be restricted to no more than nine cycles for spent fuel with burnup greater than 45 GWd/MTU. The implementation of the thermal cycling limitation for high burnup fuel (>45 GWd/MTU) is provided in NAC-UMS® Technical Specification LCO 3.1.1 and the technical bases are discussed in FSAR Appendix 12C, Technical Specification Bases for the NAC-UMS® System. This thermal cycling criterion is an acceptable alternative to ISG-11 Rev. 3 guidance. The staff evaluated the applicant’s revised thermal analysis for the new contents (high burnup PWR fuel) and agrees there is reasonable assurance that the thermal criteria will be satisfied.

In conclusion, the staff has reviewed the proposed changes and has reasonable assurance that:

- Structures, systems and components (SSCs) important to safety are described in sufficient detail in FSAR Sections 1, 2 and 4 to enable an evaluation of their thermal effectiveness. These SSCs have not changed as result of this amendment.

- As confirmed by the staff, the maximum decay heat loads were determined appropriately, and accurately reflect the burnup, cooling times, and initial enrichments specified [10 CFR 72.236(a)].

- The spent fuel cladding will be protected against degradation that leads to gross ruptures by maintaining the clad temperature below maximum allowable limits and by providing an inert environment in the cask cavity [10 CFR 72.236(a),(b) and (f)].

- The NAC-UMS® system, with the requested new contents, provides adequate heat removal capacity without active cooling systems [10 CFR 72.236(f)].

- The temperatures of the cask components and the cask pressures under normal and accident conditions were determined correctly.

- The thermal design in the FSAR is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the NAC-UMS® system, with new contents, will allow safe storage of spent fuel for a certified life of 20 years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

IV. SHIELDING EVALUATION

The objective of the shielding review is to ensure that there is adequate protection to the public and workers against direct radiation from the cask contents. The review seeks to ensure that adequate protection is maintained, consistent with any changes to the shielding features and contents, and that direct radiation exposures will be below regulatory limits for normal operating, off-normal, and design-basis accident conditions. The staff reviewed changes to the shielding design description, radiation source definition, and shielding analyses for the NAC-UMS® system proposed in this CoC amendment request. As part of this request, the applicant proposed storage of PWR fuel with an increased allowable burnup of up to 60,000 MWd/MTU.

A. Shielding Design Features and Design Criteria

A.1 Shielding and Source Term Design Criteria

The overall radiological protection design criteria are identified in the regulatory requirements of 10 CFR Part 20 and 10 CFR 72.104, 72.106(b), and 72.236(d). The applicant analyzed the NAC-UMS® system with spent fuel and hardware having the characteristics described in Section 2.1 of the proposed FSAR amendment. Although there are no numerical limits in the regulations for cask system surface dose rates, these dose rates serve as design criteria to assure there is sufficient shielding to meet radiological limits in accordance with 72.236(d).

The staff reviewed the design criteria and found them acceptable. The shielding and source term design criteria defined in the FSAR provide reasonable assurance that the NAC-UMS® system can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72. Each user is required to protect personnel from the increased dose rates in accordance with ALARA principles and the regulations of 10 CFR Part 20. Limits related to maximum decay heat, as well as the combined maximum burnup, minimum enrichment, and minimum cooling times are incorporated into the Technical Specifications of Appendix B of the CoC.

A.2 Preferential Loading Criteria

The NAC-UMS® storage system is designed to store fuel in either a standard loading pattern or a preferential loading pattern. The limitations for both loading patterns are specified in Section 2 of Appendix B to the CoC.

For the standard loading pattern, the heat load of each fuel assembly is limited to 0.958 kW. For the preferential loading pattern, the heat load of the fuel assemblies at the periphery locations is limited to 1.05 kW, and the heat load of the fuel assemblies at the basket interior locations is limited to 0.867 kW.

B. Source Specification

The design-basis source specifications for bounding calculations are presented in Section 5.2 of the FSAR amendment. A design basis fuel assembly was established for PWR and BWR fuel types. Source terms are generated for both UO_2 fuel and for fuel assembly hardware. For the shielding analysis, the Westinghouse 17x17 and GE 9x9-2L fuel assembly types are selected as the design basis PWR and BWR fuel assemblies, respectively.

As part of this amendment, the maximum allowable average burnup of PWR fuel is increased to 60,000 MWd/MTU (assembly average burnup). The design basis PWR fuel has an average burnup of 40,000 MWd/MTU, an initial enrichment of 4.2 wt% ^{235}U, and a post-irradiation cooling time of 5 years and includes source contributions from an activated burnable absorber assembly. However, to bound the neutron source produced by lower enrichment fuel which may achieve this higher burnup, the revised PWR source term is calculated with an initial enrichment of 3.7 wt% ^{235}U. This assumption produces a neutron source 30% higher than that obtained by assuming a 4.2 wt% ^{235}U initial enrichment. In order to assure that the higher burnup PWR fuel assemblies are bounded by the design basis fuel, the cooling times are extended until dose rates are below those calculated for the design basis fuel assemblies.

As part of the shielding analysis for the NAC-UMS® vertical concrete cask and the transfer cask, the applicant used the SCALE 4.3 code system. Source terms for UO_2 fuel and fuel assembly hardware were generated using the SAS2H code sequence as described in Section 5.2 of the FSAR. The 27-group neutron, 18-group gamma coupled cross section library was used as part of this SCALE package. The staff initially expressed a concern regarding the use of SCALE 4.3 for determining fuel depletion and source terms for high burnup fuel. This concern was based on the fact that there are updated versions of the SCALE code that include revisions to a number of code sequences and cross section libraries, and that neither this SCALE code nor other versions have been validated for use with PWR fuel at higher burnups.

The NAC response provided the justification for using SCALE 4.3 for the proposed high burnup fuel. The applicant included a number of references, some of which performed benchmarking calculations using SCALE 4.4 with the 44-group cross section library. The applicant also stated that NAC comparisons showed good agreement for PWR source terms generated using the two methods.

In the staff's view, care should be taken in using the SCALE 4.3 code system to calculate source terms in this manner. The references provided do not validate the use of SCALE 4.3 or SCALE 4.4 for PWR fuel with burnups up to the level proposed for this amendment (60,000 MWd/MTU assembly average burnup). Confirmatory data regarding the accuracy of SCALE to calculate source terms for PWR fuel at these higher burnups are not readily available. It is reasonable to assume that future benchmarks will be generated for fuel with higher burnups using current versions of the SCALE code, not to include SCALE 4.3.

The applicant stated that there is good agreement between SCALE 4.3 (using the 27-group library) and SCALE 4.4 (using the 44-group library) at lower burnups (≤ 40,000 MWd/MTU). However, the applicant has not provided sufficient data to suggest that they can account for uncertainties that may occur as a result of using the SCALE 4.3 code system to calculate source terms at higher burnups. Despite these potential uncertainties, the NRC staff believes that comparative results using the two code versions would not yield a substantial difference in PWR source term results at the higher burnup values proposed for this amendment (≤ 60,000 MWd/MTU), on the basis of the staff's engineering judgment and limited independent confirmatory analysis, as discussed in Section C below. Therefore, for this amendment request only, the staff finds that the PWR fuel assembly design-basis sources previously calculated for the NAC-UMS® system bound the source terms for the higher burnup PWR fuel proposed in this amendment.

B.1 Gamma Source

The design-basis fuel gamma source contains contributions from both fission products and actinides. The hardware gamma spectra contain contributions primarily from ^{60}Co due to the activation of Type 304 stainless steel with 1.2 g/kg ^{59}Co impurity and with minor contributions from ^{59}Ni and ^{58}Fe. The activated fuel assembly hardware source term was determined by multiplying the source strength from 1 kg by the total mass of steel and inconel in the plenum, upper end fitting, or lower end fitting, and then multiplying this by a regional flux activation ratio as detailed in Section 5.2.1 of the FSAR.

B.2 Neutron Source

The design-basis fuel neutron sources result from actinide spontaneous fission and from alpha-n (α, n) reactions. The isotopes ^{242}Cm and ^{244}Cm characteristically produce all but a small percentage of the spontaneous fission neutrons and (α, n) sources in PWR and BWR fuel.

C. Staff Evaluation

Sources for fuel assemblies at enrichment, burnup, and initial cooling times different from the design basis fuel (40,000 MWd/MTU), described in Section 5.2 for the FSAR, are generated within the minimum cooling time matrix determinations, outlined in Section 5.5 of the FSAR. The design basis fuel characteristics produce maximum system dose rates. Cooling times are therefore adjusted to ensure that the dose rates for any other approved fuel assembly remain bounded by the established design basis fuel assembly.

The minimum allowable cooling times for each combination of fuel type, initial enrichment, and burnup are determined by a strategy involving the use of the decay heat and dose rate data at each individual cooling time step. The minimum cooling times are determined for each limiting value for decay heat and the dose rates for the transfer cask and storage cask. The resulting minimum cooling times are rounded up to the next whole year for each combination of burnup, enrichment, and dose rates. The bounding dose rates for the storage cask and transfer cask are outlined in Table 5.5-2 of the FSAR.

The staff reviewed the source term analyses and the methodology for assuring that the design basis fuel was, in fact, bounding, as discussed in Section 5.2 of the FSAR amendment. The staff performed confirmatory analyses of selected burnup and cooling times using SAS2H and found good agreement in comparative results for the total neutron (neutrons-second-assembly) and total gamma (MeV-second-assembly) sources using SCALE 5.1. Staff confirmatory results for higher burnups (≤ 60,000 MWd/MTU) of PWR fuel showed higher neutron sources, but smaller overall source totals for a cooling time of 18 years. The gamma sources for this cooling time were on the order of 10^{15} versus 10^{8} for the neutron sources. The confirmatory calculations also showed a reasonable margin in gamma source term between the design-basis PWR fuel and higher burnup PWR fuel chosen for the staff’s confirmatory analysis.

The applicant established a design basis fuel assembly (40,000 MWd/MTU) that is within referenced benchmark data, and employed the method of increasing the cooling time to ensure that dose rates remain below those established as the bounding dose rates, as indicated in Table 5.5-2 of the FSAR. Although the applicant did not account for possible uncertainties involved with using the SCALE 4.3 code to calculate source terms at the higher burnups, some degree of margin exists between the PWR design-basis fuel assembly source terms and the higher burnup PWR fuel assemblies proposed in this amendment. The staff has reasonable confidence that the methodology and operational practices to limit dose rates and decay heat to those specified in the FSAR are acceptable. However, it should be noted that the methodology used for the shielding evaluation as part of this amendment was only deemed acceptable in conjunction with other factors brought out as part of this review. The use of SCALE 4.3 as a means of determining fuel depletion and source terms for high burnup fuel for other amendment requests involving this or other cask systems will be considered by the staff on a case-by-case basis.

The applicant also proposed to revise a note on FSAR Drawing 790-585 to make the use of the structural lid and shield lid threaded plugs and dowel pins optional. The applicant’s justification for this change is that the plugs and pins were intended to replace the shielding material that was removed from the holes, but the dose incurred by personnel installing/removing the plugs will exceed the anticipated dose reduction. The applicant states that the regulatory defined and calculated site boundary doses are essentially unaffected. The staff finds the applicant’s justification for this change acceptable.

The staff concludes that the NAC-UMS® system is designed and can be operated in compliance with the radiation protection requirements of 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the NAC-UMS® system will provide safe storage of spent fuel. This finding is based on a review of the material presented, confirmatory analysis performed by NRC staff, and acceptable engineering practices.

V. CRITICALITY EVALUATION

The objective of the criticality review is to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered for the proposed changes to the NAC-UMS® Universal Storage System such that the cask system continues to meet the criticality requirements of 10 CFR Part 72. These requirements include: 10 CFR 72.124(a), 72.124(b), 72.236(c), and 72.236(g). The applicant did not make changes to the criticality analysis in its proposed amendment to the NAC-UMS® system.

The staff reviewed the changes proposed to other areas of the FSAR, Certificate of Compliance, and Technical Specifications (TS) to ensure that any impacts on criticality were appropriately considered. The staff noted that the applicant proposed a change to Section B 2.1.2 of Appendix B (TS) to the CoC. The change was to explicitly limit the authority for preferential loading to Maine Yankee (MY) high burnup fuel only. The staff interprets the TS such that the allowable contents and configuration descriptions given in Section B 2.1.2 apply only to MY fuel; therefore, the need for this statement was unclear to the staff and raised concerns that the contents and configurations descriptions in this Section of Appendix B could be interpreted to apply to non-MY fuel as well as MY fuel. Such an interpretation is not consistent with the staff's interpretation and is not supported by the current criticality analysis. The applicant, in response to staff questions, clarified that the allowable contents and configurations descriptions given in Section B 2.1.2 apply solely to MY fuel, as indicated by the title of this TS section, and removed the proposed change. Thus, the staff finds that there were not any changes proposed to the TS, CoC or FSAR that affect system reactivity.

The staff also identified two other items that, while not affected by the proposed amendment, necessitated further review. These two items are the definitions of intact and damaged fuel and the lack of a minimum basket dimension limit as part of Section B 3.2, Appendix B to the CoC, "Design Features Important for Criticality Control." These two items are important for controlling criticality in the cask and ensure that key assumptions in the criticality analysis remain valid for the cask system.

The first item is the definitions of intact and damaged fuel. The definition of intact fuel previously included the requirement that missing fuel rods shall be replaced by solid filler rods that displace an amount of water equal to that displaced by the original rods. In a previous amendment, this portion of the definition was removed. In the same amendment, MY fuel of various configurations was added to the approved cask contents. Some of the MY assemblies were considered structurally intact, but they were missing one or more fuel rods; thus, they could not be considered intact under the then current TS definition of intact fuel. So, the definition was modified to allow these MY assemblies to be classified as intact and a criticality analysis was performed to support this modification with respect to MY fuel only. However, the change in the definition allows for classification of non-MY fuel assemblies that are missing one or more fuel rods not replaced by solid filler rods to be classified as intact fuel as well, a condition which is not supported by the current criticality analysis. The analysis for non-MY fuel continues to rely upon the key assumption, or condition, that empty fuel rod positions are filled with solid filler rods (see FSAR page 6.2-1, last paragraph). Thus, to prevent this condition, the applicant further proposed a modification to the TS definitions so that non-MY fuel assemblies that are missing fuel rods which are not replaced by solid filler rods (of equal or greater volume as the original rods) are classified as damaged fuel. These changes are incorporated with those definition changes in response to other staff questions (see the materials evaluation of this SER). These last changes further modified the TS definitions to include undamaged fuel, with associated modifications also made to the definitions of damaged and intact fuel. The staff reviewed the proposed changes with respect to criticality and finds them to be acceptable.

The second item is the lack of an appropriate minimum basket dimension for the pressurized water reactor (PWR) and boiling water reactor (BWR) fuel baskets. The basket geometry is a key mechanism for controlling system reactivity in the cask (see FSAR pages 6.1-2 and 6.1-3). The applicant's analysis indicates that system reactivity increases as fuel assemblies are brought closer together, and, in the case of the PWR basket, the width of the flux traps is reduced. The applicant's analysis also assumes a minimum dimension for the flux trap width in the PWR basket and a minimum separation between fuel tubes in the BWR basket. The staff recognizes that the applicant's criticality analysis accounts for design tolerances and fuel positioned in the fuel tubes in a way that maximizes system reactivity. Thus, the staff finds there is reasonable assurance that the cask system with the current PWR and BWR basket designs will remain subcritical. However, the staff finds it important to ensure that design parameters that are significant to criticality safety control are included in the Technical Specifications; the minimum flux trap thickness (web thickness) is one such parameter.

In response to staff's questions, the applicant proposed minimum flux trap (web thickness) values for both the PWR and BWR baskets in Section B 3.2.1 of the Technical Specifications. These values capture the minimum values that are possible with the current basket design, including tolerances. However, the staff questioned whether the proposed TS modification was consistent with, or was captured by, the criticality analysis, since the analysis does not consider all flux traps to be at their minimum dimension concurrently. To address this question, the applicant modified the proposed TS to refer to two figures (proposed Figures B2-1 and B2-2) to illustrate the minimum flux trap/web thickness for the PWR and BWR baskets. The applicant added a note to the figures to allow for rare instances when a fabrication error may occur that could result in some basket disk openings that may be slightly outside the specifications given in the figures. The staff recognizes that there may be instances when such a fabrication error could occur; however, it is staff's understanding that such an error is uncommon and should not affect a large number of disk openings in the same basket disk, and therefore, is highly unlikely to affect the criticality design/analysis. Therefore, with the modification to the note of proposed TS Figures B2-1 and B2-2 to state that variations from the dimensions in the figures resulting from fabrication errors will be limited, as specified, the staff finds the proposed flux trap TS to be acceptable.

The staff has reviewed the applicant's proposed amendment and finds, as discussed in this section of this SER, that there is reasonable assurance that the cask system will remain subcritical under all credible conditions and that the criticality design features meet the regulatory requirements of 10 CFR Part 72.

VI. CONFINEMENT EVALUATION

The objective of the confinement review is to ensure that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that otherwise might lead to gross ruptures.

NAC's initial submittal for amendment to the CoC for the NAC-UMS® system, dated September 22, 2006, requested the removal of the helium leakage test of the canister shield lid welds, based on implementing the requirements of ISG-18 for confinement closure weld qualification. In response to an RAI on the UMS operating procedures requesting clarification on how the remaining vent and drain port cover weld leakage rate tests comply with ANSI N14.5, the licensee revised the confinement section, operating procedures, and TS to revert to the previously approved leakage rate test criteria, and removed any references to ISG-18. As a result, the proposed amendment has no changes to the confinement section of the FSAR.

VII. MATERIALS EVALUATION

The objective of the materials evaluation, as described in ISG-15, is to ensure that materials selected for each “important to safety” component of the dry cask storage system are adequate to perform the safety function(s) required of that component, under normal, off-normal and accident conditions. In addition, the materials evaluation is also intended to confirm that the spent fuel cladding is protected from gross rupture and from conditions that could lead to fuel redistribution.

The amendment to the CoC for the NAC-UMS® system requests three changes that could have materials implications; 1) removal of helium leak test for the canister shield weld to canister shell weld, 2) elimination of the Charpy V-notch testing on the BWR support disk, and 3) adding high burnup PWR fuel to the approved contents list. In the course of this review, the definition of intact fuel was also scrutinized in light of issues that have recently arisen with detecting damaged fuel that might be loaded in this cask.

With respect to the removal of helium leak test for the canister shield weld to canister shell weld, the staff previously indicated that the removal of the helium leak test on the said weld would be acceptable, if the applicant specified that the weld would have three passes. The applicant declined to commit to this, and subsequently withdrew this part of the amendment request. Thus the requirement will remain unchanged.

The staff finds acceptable the removal of the Charpy V-notch test on the BWR support disk. The BWR support disk is 5/8 inch thick; therefore, it meets the ASME Code Section III, Subsection NG-2311(a)(1) requirement for exemption from the Charpy test. The deletion of the related note on FSAR Drawing 790-573 is also acceptable.

The staff questioned the applicant on whether the cooling time for the high burnup fuel was sufficient to meet the heat rejection requirements for the system, and whether the stated amount of fission gas corresponded to the 60 GWd/MTU limit. NAC indicated that the cooling is bounded by the FSAR calculations and the fission gas generation was appropriate for the 60 GWd/MTU limit. The staff did not review the previous analysis methodology except for verifying that the fuel temperatures meet the acceptance criteria for fuel cladding. To preclude fuel cladding degradation, the applicant limited the maximum cladding temperature under normal conditions of storage and short-term canister transfer operations to 752°F (400°C), and for off-normal and accident events, to 1058°F (570°C), consistent with ISG-11, Revision 3. The applicant also proposed critieria to limit thermal cycling during system drying operations. For spent fuel with burnup greater than 45 GWd/MTU, cladding temperature fluctuations that exceed 117°F (65°C) are restricted to no more than nine cycles. The staff found this alternative thermal cycling criterion to ISG-11 Revision 3 guidance acceptable. The staff evaluated the applicant's revised thermal analysis for the new contents (high burnup PWR fuel) and agrees there is reasonable assurance that the criteria for limiting thermal cycling will be satisfied, and that the fuel cladding will retain its function under all conditions of storage. Since other materials properties have been adjusted for the higher burnup limit, and no new compatibility issues arise due to the increased burnup, the staff finds it acceptable to include high burnup PWR fuel as approved contents for storage, with respect to materials considerations.

A recent NRC inspection revealed that at least one utility has revised the definitions of intact and damaged fuel for loading into this cask system using the 72.48 process. This change introduced the possibility for assembly damage different from what was previously accepted by the staff, and also allowed for the use of visual examination as a tool for detecting damaged fuel rods. Consequently, NAC was asked to revise the definitions in the FSAR and CoC (TS) to reflect the functional characterization of spent fuel as discussed in ISG-1, Revision 2, and to describe the process to be used for determining fuel condition. These revisions have been incorporated by NAC in its submittals dated September 6 and September 26, 2007, and April 23, 2008, and the staff approves the corresponding changes.

VIII. CERTIFICATE OF COMPLIANCE AND TECHNICAL SPECIFICATION CHANGES

The objective of the review of the proposed changes to the Certificate of Compliance (CoC) and associated Technical Specifications (TS) for the NAC-UMS® system is to determine if the proposed changes impose or revise CoC conditons or TS requirements to appropriately reflect the design modifications and revised analyses of the amendment request. This review focused on evaluating whether the Conditions and TS had been revised to ensure that all safety limits and regulations were met. In addition to those proposed changes to the TS and FSAR to incorporate the high burnup fuel (up to 60 GWd/MTU assembly average burnup), the applicant proposed further changes to the NAC-UMS® Certificate of Compliance, the associated TS, and the FSAR.

In the CoC, the applicant proposed eliminating the requirement for the use of tamper-indicating seals on the concrete cask lid bolts. The use of the seals is currently stated in the NAC-UMS® CoC section entitled, "Approved Spent Fuel Storage Cask - Description." In the staff's view, this language in the CoC was intended to describe the cask system, not to impose a new requirement in this instance. Tamper indication on storage cask components is not required by 10 CFR 72. Therefore, the staff agrees that the description of these tamper-indicating seals can be removed from the CoC. A related note on FSAR Drawing 790-590 will be added to make the use of the seals optional.

The applicant proposed several additional changes to the NAC-UMS® TS, beyond those addressed in earlier sections of this SER. Technical Specification B 2.1, states that if any Fuel Specification or Loading Conditions of this section are violated, the following actions shall be completed:

- The affected fuel assemblies shall be placed in a safe condition.
- Within 24 hours, notify the NRC Operations Center.
- Within 30 days, submit a special report that describes the cause of the violation and actions taken to restore or demonstrate compliance and prevent recurrence.

The applicant proposed that the written reporting requirement for TS B 2.1 be revised to 60 days, instead of 30 days. The justification for the change is that it would make the TS requirement consistent with a similar requirement in 10 CFR 72.75(g). The staff notes that 10 CFR 72.75 does not explicitly include fuel misloading or related events among those events requiring reporting to NRC; thus, the TS requirement was imposed to ensure that NRC will be promptly notified of such an event. The staff also notes that 10 CFR 72.75(g) provides greater detail on the required content of the written report than in proposed TS B 2.1. The 60-day written reporting requirement in 10 CFR 72.75(g) covers non-emergency events of varying severity, including those requiring 4-hour, 8-hour and 24-hour notification of NRC. The staff agrees that the same 60-day time period for submittal of a written report for a violation of fuel specifications or loading conditions is appropriate, and NAC has proposed to further revise TS B.2.1 to specify that the written report will meet the applicable requirements of 10 CFR 72.75(g). Therefore, this change is acceptable.

The applicant proposed editorial corrections to TS Section A 5.6, “NAC-UMS® SYSTEM Transport Evaluation Program.” As described in NAC’s letter dated May, 8, 2007, an editorial error occurred in the processing of Amendment 4 when cross-references to TS Section B 3.4.1(6) were inadvertently left in TS A 5.6, when B.3.4.1(6) was removed. Additionally, TS Section A 5.6 directed the user to refer to the cask tip-over analyses in the FSAR, in order to ensure that the site-specific transport and pad surfaces are appropriately bounded by the FSAR analyses. NAC proposes to remove those references and instead refer to the FSAR analysis of the 24-inch vertical cask drop accident, which is more limiting than the cask tipover analyses. The staff finds these editorial changes acceptable.

The applicant proposed an editorial clarification to TS Table B3-1, “List of ASME Code exceptions for the NAC-UMS® SYSTEM.” The change deletes the sentence, “Leaktightness of the CANISTER is verified by testing,” for the component identified as “Canister Shield Lid and Structural Lid Welds.” This is to clarify that the shield lid to canister shell weld, which forms the primary confinement boundary, is subject to a leak test, but the structural lid to canister shell weld is not; it is subject to an ultrasonic or liquid penetrant examination. The staff finds this change acceptable.

IX. CONCLUSION

The NRC staff has reviewed the amendment application, as supplemented, for the NAC-UMS® system. The staff considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices in reaching its conclusion. Only the requested changes were addressed in the scope of this SER. The Certificate of Compliance and the associated TS, and the FSAR have been revised to reflect the changes requested by NAC. Those changes consist of: 1) revising the TS for allowable contents to include high burnup PWR fuel up to 60 GWd/MTU, and the related FSAR descriptions and analyses; 2) eliminating the CoC requirement for the use of a tamper-indicating seal on the concrete cask lid bolts; 3) revising the Technical Specification Section B2.1 written reporting requirement from 30 to 60 days; 4) eliminating the requirement in the FSAR for Charpy V-notch impact testing of the 0.625-inch nominal thickness BWR support disk material (SA 533, Type B, Class 2); 5) making the use of the structural lid and shield lid threaded plugs and dowel pins optional, as specified in the FSAR drawings; 6) revising the TS definitions of intact and damaged fuel to conform with the NRC guidance of ISG-1, Revision 2; and 7) revising the TS requirements for limiting potential thermal cycling of the fuel cladding, for consistency with the staff's position in ISG-11, Revision 3.

Based on the statements and representations contained in the application, as supplemented, the staff concludes that these changes do not affect the ability of the NAC-UMS® Universal Storage System to meet the requirements of 10 CFR Part 72, and therefore, the proposed changes are acceptable.

Issued with Certificate of Compliance No. 1015, Amendment No. 5 on **DRAFT** .