November 24, 2004

Mr. Thomas C. Thompson, Director
Licensing Engineering
NAC International
3930 East Jones Bridge Road
Suite 200
Norcross, GA 30092

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9270, REVISION 1, FOR MODEL NO. NAC-UMS PACKAGE (TAC NO. L23726)

Dear Mr. Thompson:

As requested by your application dated March 31, 2004, as supplemented June 11 and September 28, 2004, enclosed is Certificate of Compliance No. 9270, Revision No. 1, for your Model No. NAC-UMS package [USA/9270/B(U)F-85]. In addition to the changes you requested, we made three changes to Certificate of Compliance No. 9270. These changes were necessary for continued conformance of Certificate of Compliance No. 9270 with revised 10 CFR Part 71. This recent revision of 10 CFR Part 71 became effective on October 1, 2004. Also, we made several minor editorial corrections. This certificate supersedes, in its entirety, Certificate of Compliance No. 9270, Revision No. 0, dated October 31, 2002. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

NAC International has been registered as a user of the package under the general license provisions of 10 CFR §71.17 or 49 CFR §173.471. The approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR §173.471.

If you have any questions regarding this certificate, please contact me or Stewart W. Brown of my staff at (301) 415-8500.

Sincerely,
/RA/
John D. Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9270

Enclosures:
1. Certificate of Compliance No. 9270, Rev. No 1
2. Safety Evaluation Report

cc: R. Boyle, Department of Transportation
J. Shuler, Department of Energy
RAMCERTS

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NAME	SBrown		MDeBose		JSmith		DTang		ADias	
DATE	11/16/04		11/12/04		11/16/04		11/15/04		11/15/04	

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SAFETY EVALUATION REPORT

Docket No. 71-9270
Model No. NAC-UMS Package
Certificate of Compliance No. 9270
Revision No. 1

SUMMARY

By application dated March 31, 2004, as supplemented June 11 and September 28, 2004, NAC International (the applicant) requested an amendment to Certificate of Compliance (CoC) No. 9270, for Model No. NAC-UMS package. The applicant requested that CoC No. 9270 be revised to: (1) replace the words “Zircaloy” and “Zircalloy” with either the words “zirconium alloy” or “zirconium alloy type,” (2) include newly and revised drawings, and (3) replace the word “BORAL” with the words ”neutron absorber.” In addition, the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) made three changes to CoC No. 9270 to address changes made to Title 10 of the Code of Federal Regulations (10 CFR) Part 71 that took effect on October 1, 2004. The staff determined that these changes to CoC No. 9270 were necessary to ensure continued compatibility with 10 CFR Part 71.

Based on the statements and representations in the application, the staff agrees that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

EVALUATION

1.0 GENERAL INFORMATION

By application dated March 31, 2004, as supplemented June 11 and September 28, 2004, the applicant requested an amendment to CoC No. 9270 to: (1) replace the words “Zircaloy” and “Zircalloy” with either the words “zirconium alloy” or “zirconium alloy type,” (2) include new and revised drawings, and (3) replace the word “BORAL” with the words “neutron absorber.” In support of this request, the applicant provided Revision 1 of its Safety Analysis Report (SAR). In addition, the staff made three changes to CoC No. 9270 to address changes made to 10 CFR Part 71 that took effect on October 1, 2004. The staff determined that these changes to CoC No. 9270 were necessary to ensure continued compatibility with 10 CFR Part 71.

The applicant requested that the words “Zircaloy” and “Zircalloy” be replaced with either the words “zirconium alloy” or “zirconium alloy type.” Zirconium alloy includes Zircaloy-2, Zircaloy-4, ZIRLO, and M-5. There are minor variations of alloy composition. However, these minor variations have no significant effect on the performance of the cladding material. Therefore, the staff determined that replacing the words “Zircaloy” and “Zircalloy” with either the words “zirconium alloy” or “zirconium alloy type” is acceptable.

Drawings

The applicant submitted the following drawings in support of this amendment request:

790-500, Rev. 4 (Sheets 1-2)	Assembly, Universal Transport Cask, Overpack
790-505, Rev. 2 (Sheets 1-2)	Lifting Trunnion
790-571, Rev. 3	Bottom Weldment, Fuel Basket, 56 Element BWR
790-575, Rev. 10 (Sheets 1-2)	BWR Fuel Tube
790-581, Rev. 9 (Sheets 1-2)	PWR Fuel Tube
790-582, Rev. 11 (Sheets 1-2)	Shell Weldment Canister
790-583, Rev. 7	Assembly, Drain Tube, Canister
790-584, Rev. 17 (Sheets 1-3)	Detail, Canister
790-585, Rev. 16 (Sheets 1-3)	Transportable Storage Canister
790-587, Rev. 1	Spacer Shim, Canister
790-591, Rev. 6 (Sheets 1-2)	Bottom Weldment, Fuel Basket, 24 Element PWR
790-592, Rev. 8	Top Weldment, Fuel Basket, 24 Element PWR
790-593, Rev. 7 (Sheets 1-2)	Support Disc and Misc. Basket Details, 24 Element PWR
790-595, Rev. 9 (Sheets 1-2)	Fuel Basket Assembly, 24 Element PWR
790-605, Rev. 11 (Sheets 1-2)	BWR Fuel Tube, Over-Sized Fuel
790-611, Rev. 5 (Sheets 1-2)	GTCC Waste Basket, Maine Yankee
790-612, Rev. 8 (Sheets 1-2)	GTCC Waste Canister, Maine Yankee
412-501, Rev. 4 (Sheets 1-2)	Spent Fuel Can Assembly, Maine Yankee
412-502, Rev. 5 (Sheets 1-6)	Fuel Can Details, Maine Yankee

The staff has concluded that the new and revised drawings will not affect the ability of the Model NAC-UMS package to meet the requirements of 10 CFR Part 71.

The applicant requested that the word “BORAL” be replaced with the words “neutron absorber” in CoC No. 9270. This change would allow use of a consistent terminology between CoC No. 9270 and Model No. NAC-UMS package design drawings for a specific component. In design drawings 790-575, Revision 10; 790-581, Revision 9; and 790-605, Revision 11 one of the components is specified as the “neutron absorber.” Further, in these same drawings the material specified for fabrication of the “neutron absorber is “BORAL.” Should the applicant want to use a material other than “BORAL” to fabricate the “neutron absorber” for the Model No. NAC-UMS package an amendment to CoC No. 9270 would be required. Thus, the staff has determined that replacing the word “BORAL” with the words “neutron absorber” in CoC No. 9270 is acceptable.

2.0 STRUCTURAL EVALUATION

This section evaluates the structural performance of the Model NAC-UMS package for the proposed modified design features.

In Drawing 790-584, Revision 17, “Details, Canister NAC-UMS,” the applicant changed the depth of the Vee-groove partial penetration weld from 0.5 inch nominal to 0.48 $\pm$ 0.12 inches for the shield lid-to-shell weld. Similarly for the structural lid-to-shell weld, the Vee-groove depth was reduced from 1.1 inches to 1 inch. The applicant evaluated effects of the weld size changes by using the new weld dimensions in a finite element re-analysis of the Transportable Storage Canister. In the applicant’s letter dated September 28, 2004, which provided responses to the staff’s request for additional information, the applicant stated that, for the end drop orientation, the load through the canister shell does not change as result of the weld reduction. For side drop orientation, a theoretical analysis of the attenuation rate, in the direction along the canister axis, for the shell bending moment is also presented to demonstrate that only a limited region at close proximity to the welds needed to be considered for canister stress margin reevaluation. As reported in the revised stress summary tables, due to the small reduction of weld sizes, all calculated stresses remain below the allowables.

In Drawing 401-501, Revision 4, the applicant introduces an alternative Maine Yankee fuel can design. The alternative fuel can design is essentially identical to the approved Maine Yankee fuel can design. The length of both fuel cans is 162.8 inches. However, the alternative fuel can design has a reduced cross-sectional cavity size. The cross-sectional cavity size was reduced from a 8.52 inch square to a 8.32 inch square. This reduction would allow loading of the alternative fuel can in the corner fuel positions of a basket in which the bottom weldment is not enlarged. Compared to the approved fuel can design, the alternative fuel can design has a slightly smaller cross-sectional area and section modulus in resisting the axial force of an end drop and the bending moment of a side drop, respectively. As a result, Section 2.11.1.1.1 of the application considered sectional properties of the alternative fuel can design, which are governing, to perform a bounding structural re-analysis to demonstrate adequate structural performance for both Maine Yankee fuel can designs.

The applicant in its letter dated March 31, 2004, submitted a number of updated drawings. Attachment 1 to that letter lists the drawing changes with justification for the changes. The staff has reviewed these changes and determined that they are generally administrative and editorial in nature. In addition, the staff determined that minor revisions of non-safety related design/fabrication features were also incorporated in the updated drawings.

The staff has concluded, based on its review, that the slightly modified Model NAC-UMS package is structurally capable of meeting the 10 CFR Part 71 requirements.

3.0 THERMAL EVALUATION

The applicant demonstrated that the proposed alternative fuel can design for Maine Yankee fuel does not introduce any change to the results and conclusions presented in the original application. Therefore, the staff concludes that the thermal performance provided by the Model NAC-UMS package continues to meet the requirements of 10 CFR Part 71.

4.0 CONTAINMENT

The changes made to the SAR do not impact the previously approved containment analysis. However, during the review, the staff identified that the Model NAC-UMS package was initially certified before the issuance of Revision 1 of Interim Staff Guidance (ISG)-1. The staff suggested that NAC consider updating its definition of damaged fuel to be consistent with current staff guidance for damaged fuel. Revision 1 of ISG-1 states:

> Damaged fuel - Spent nuclear fuel is considered damaged for storage or transportation purposes if it manifests any of the following conditions that result in either compromise of cladding confinement integrity or rearrangement (reconfiguration) of fuel bundle geometry:
>
> 1. The fuel contains known or suspected cladding defects greater than a pinhole leak or hairline crack that have the potential for release of significant amounts of fuel particles into the cask.
>
> 2. The fuel assembly:
>
>   a. Is damaged in such a manner as to impair its structural integrity;
>
>   b. Has missing or displaced structural components such as grid spacers;
>
>   c. Is missing fuel pins which have not been replaced by dummy rods which displace a volume equal to or greater than the original fuel rod;
>
>   d. Cannot be handled using normal (i.e., crane and grapple) handling methods. (Exception: fuel assemblies with repaired lifting bails, support caps, or support tubes, etc., which permit normal handling may be classified as intact. See later discussion.)
>
> 3. The fuel is no longer in the form of an intact fuel bundle and consists of, or contains, debris such as loose fuel pellets, rod segments, etc.

NAC revised its SAR to incorporate this definition of damaged spent nuclear fuel, except for item 2.c, which refers to missing fuel pins that have not been replaced.

The staff concludes that the slightly modified definition of damaged fuel is adequate for the previously approved contents (Maine Yankee fuel) which was previously evaluated by the staff (CoC No. 9270, revision 0, Docket 71-9270, dated October 31, 2002).

However, the staff recommends that the applicant consider fully adopting the staff's current guidance on damaged fuel should the applicant apply for authorization to load contents other than Maine Yankee spent fuel in the future.

The staff notes that while the assumed failure rates for high burnup fuel rods are very conservative, they are based on an outdated ISG. ISG-15, "Materials Evaluation," Section X.5.4.2, "High Burnup Fuel," has been superceded by Revision 3 of ISG-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel." The staff encourages the applicant to refer to this guidance in the future.

The staff has reviewed Revision 1 to the SAR for the Model NAC-UMS package. Based on the statements and representations contained in the SAR and the conditions specified in the CoC No. 9270, the staff concludes that the package meets the containment performance requirements of 10 CFR Part 71.

5.0 SHIELDING EVALUATION

There were no revisions to Chapter 5, "Shielding Evaluation," of the SAR that affected the level of radiation protection provided by the Model NAC-UMS package. Therefore, the staff concludes that the radiation protection provided by Model NAC-UMS package continues to meet the requirements of 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

NRC staff performed a criticality safety review of the proposed amendment for the Model NAC-UMS package to incorporate updated information. This information affecting the criticality safety of the package included removing the restrictions of which types of fuel assemblies can be loaded into a canister, replacing the term "BORAL" with the generic term "neutron absorber," and updating the Maine Yankee fuel can description and performing additional calculations to support the design configuration. The staff evaluated the proposed updates based on information provided in the amended CoC, the updated SAR, and NAC's responses to the NRC's request for additional information.

The Model NAC-UMS package is designed to transport up to 24 intact PWR fuel assemblies with an initial enrichment of 4.2 weight-percent (wt.%) ^{235}U or up to 56 intact BWR fuel assemblies with an initial enrichment of 4.0 wt.% ^{235}U. The various types of fuel assemblies are sorted into classes by length and cross sections. This sorting results in three classes of PWR fuel assemblies (Classes 1, 2, and 3) and two classes of BWR fuel assemblies (Classes 4 and 5). All classes of assemblies evaluated include the fuel basket in the calculations. The packaging design features relied on to maintain subcriticality are the basket geometry and the fixed neutron poisons contained in the basket.

The PWR basket utilizes the flux trap principle to maintain criticality control. The fuel assemblies are surrounded by a neutron absorber that contains at least 0.025 grams $^{10}B/cm^2$ and are separated by an air gap that is assumed flooded during hypothetical accident conditions. Neutrons escaping one fuel assembly are moderated by the gap and absorbed by the neutron poison. Flux trap spacing is maintained by stainless steel support disks that are part of the fuel basket and serve to separate individual fuel assembly tubes. Different basket configurations are used for each of the three fuel basket classes.

The BWR basket utilizes a water gap and a single neutron absorbing sheet containing at least 0.010 grams $^{10}B/cm^2$. Although this type of arrangement is not a flux trap per se, this configuration also absorbs thermal neutrons in the poison sheet during accident conditions. Since the BWR assemblies contain a smaller amount of fissile material compared to the PWR assemblies, this single poison sheet provides adequate criticality control. Spacing is maintained by carbon steel disks in two configurations based on the fuel class and the 95 neutron absorber sheets used in the basket are able to ensure that at least one poison sheet is between any two adjacent fuel tubes.

This amendment update replaces the term "BORAL" with the more generic term "neutron absorber" throughout the SAR. This substitution does not adversely impact the evaluation of the Model NAC-UMS package since the main criteria for a neutron poison is the amount of ^{10}B present in the absorber. The concentration levels of boron used as neutron absorbers for all classes of fuel assemblies has not changed from the values used in the previous application.

Analyses were performed by the applicant to encompass credible fuel configurations, including normal and hypothetical accident conditions to ensure that effective multiplication factor (k_{eff}) was below the Upper Subcritical Limit (USL) for all analyzed configurations. The fuel may contain various enrichments of ^{235}U (up to a maximum of 4.2 wt% for PWR fuel and 4.0 wt% for BWR fuel) as specified by fuel type in the CoC and is considered fresh (i.e., no burnup credit is taken) for all criticality calculations to maximize the potential reactivity of the fuel. The licensee used the SCALE 4.3 criticality sequence computer code package using the 27GROUPNDF4 cross-section library for fissile and shielding media in their calculations and used MONK8a to evaluate the change in reactivity as a result of the Model NAC-UMS package top end drop accident scenario.

The applicant determined the most reactive PWR and BWR fuel assemblies in the corresponding basket configuration to be the Westinghouse 17 x 17 OFA for PWR assemblies and the Exxon/ANF 9 x 9 with 79 fuel rods for BWR assemblies. These assemblies were taken as bounding for all fuel types in their respective classes (i.e., Classes 1, 2, and 3 for PWR fuel assemblies and Classes 4 and 5 for BWR fuel assemblies).

Additional information was provided by the applicant regarding the top end impact evaluation and the exposed fuel height methodology and additional clarifications were submitted. The originally analyzed exposed fuel heights were found to be conservative and no revised criticality calculations were necessary.

Damaged fuel assemblies may be placed in two configurations of the Maine Yankee fuel can prior to loading in a basket. Any fuel debris must be loaded in a rod or tube structure that is subsequently loaded into one of the fuel cans and is limited to the mass equivalent of an intact fuel rod of an undamaged fuel assembly. These damaged fuel cans may only be placed in one of the four corner positions of a basket. As shown by the applicant's analysis, assemblies with up to 176 damaged rods and consolidated assemblies with up to 289 rods are allowable contents and are equivalent or less reactive than a cask loaded with bounding PWR fuel.

Table 6.1-1 of the amended SAR contains a summary of the final analysis results for the most reactive PWR and BWR assemblies under both Normal Conditions of Transport and Hypothetical Accident Conditions. These results are for a single package and for arrays of damaged and undamaged packages, as required by 10 CFR 71.55 and 71.59. The maximum k_{eff} for each condition as calculated by the applicant is summarized in the table below and indicates that all of the results are below the USL for each code type.

Calculated Maximum k_{eff}				
Condition	**Code Used**	**keff + 2σ**		**USL**
		PWR	**BWR**	
Single Package, Flooded 10 CFR 71.55(b), (d), and (e)	SCALE4.3	0.9265	0.9071	0.9361
Infinite Array of Undamaged Packages, Dry 10 CFR 71.59(a)(1)	SCALE4.3	0.4002	0.4021	0.9361
Infinite Array of Damaged Packages, Flooded 10 CFR 71.59(a)(2)	MONK8a	0.9351	0.9373	0.9426

As can be seen from the results, the proposed changes in this amendment request do not affect the calculated k_{eff} in any way, and the results are identical to the original approval for this package.

NRC staff performed confirmatory calculations of the normal and the most reactive damaged scenarios of the bounding configurations of the Model NAC-UMS package using the KENO V.a code and the 44GROUPNDF5 cross section set in the SCALE4.4a system of codes. The results of these confirmatory calculations were consistent with those performed by the licensee. In all instances, the calculated k_{eff} was found to be below the USLs for each fuel type when the code biases and uncertainties were added, ensuring an adequate margin of safety.

Based on NRC staff verification of adequate system modeling by the licensee and that the acceptance standard of a maximum k_{eff} + 2σ # 0.95 was maintained for all analyzed scenarios, the analyses supporting the updates specified in the amendment were considered acceptable and the staff concludes that the package meets the criticality performance requirements of 10 CFR Part 71.

7.0 OPERATIONAL PROCEDURES

There were no revisions to Chapter 7, “Operational Procedures,” of the SAR that affected the effectiveness of the operating procedures for package loading, unloading, preparation of an empty package for transport, and preparation of a package that was used in spent fuel storage. Therefore, the staff concludes that Model NAC-UMS package continues to meet the requirements of 10 CFR Part 71.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

There were no revisions to Chapter 8, “Acceptance Tests and Maintenance Program,” of the SAR. Therefore, the staff concludes that the Acceptance Tests and Maintenance Program continue to meet the requirements of 10 CFR Part 71.

Revised 10 CFR Part 71

On January 26, 2004, NRC published its final rule revising 10 CFR Part 71, “Packing and Transportation of Radioactive Material.” NRC revised Part 71 to address compatibility with the International Atomic Energy Agency’s transportation safety standards, “Regulation of the Safe Transport of Radioactive Material” (TS-R-1) and other transportation safety issues. The revised 10 CFR Part 71 final rule was published in the *Federal Register* (69 FR 3698). This rule became effective on October 1, 2004.

The NRC staff has determined that as a result of changes made to 10 CFR Part 71 several changes to CoC No. 9270 were necessary to ensure continued compatibility with the revised regulation:

1. Condition 5(b)(1)(iv), page 18 of 20 - Revised wording from:

   > “Maine Yankee GTCC waste consists of solid, irradiated, and contaminated hardware and solid, particulate debris or filter media, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.53.”

   to:

   > “Maine Yankee GTCC waste consists of solid, irradiated, and contaminated hardware and solid, particulate debris or filter media, provided the quantity of fissile material does not exceed a Type A quantity and does not exceed the mass limits of 10 CFR 71.15.”

   Section 71.53 provided the methods of determining the mass limits for exemption from the fissile material package standards of Sections 71.55 and 71.59. Because these mass limits are so low, not exceeding them provides a means for controlling the potential of a criticality event associated with packaging of greater than class C (GTCC) waste. Effective October 1, 2004, Section 71.53 was deleted and a new Section 71.15, provides the methods of determining the mass limits for exemption from the fissile material package standards of Sections 71.55 and 71.59. Although the methods for determining mass limits provided in Section 71.15 are different from those provided in Section 71.53, the resulting mass limits are still sufficiently low to preclude a criticality event.

2. Condition 5(c), page 19 of 20 - Revised wording from:

   "Transport Index for Criticality Control (Criticality Safety Index)
   Minimum transport index to be shown on
   label for nuclear criticality control: 0.0"

   to:

   "Criticality Safety Index: 0.0"

   Effective October 1, 2004, packages containing fissile material were required to be labeled with a "Criticality Safety Index" rather than a "Transport Index." This change was necessary to ensure continued compatibility with 10 CFR Part 71.

3. Condition 8, page 19 of 20 - Revised wording from:

   "The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12."

   to:

   "The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17."

   As part of the 10 CFR Part 71 revision certain sections were renumbered with no substantial change to the renumbered section. This change was necessary to address the renumbering of Section 71.12 to Section 71.17, that became effective on October 1, 2004.

CONCLUSION

CoC No. 9270 has been revised to: (1) replace the words "Zircaloy" and "Zircalloy" with either the words "zirconium alloy" or "zirconium alloy type," (2) include newly and revised drawings, and (3) replace the word "BORAL" with the words "neutron absorber." Based on the statements and representations in the application, the staff agrees that the changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71. In addition, CoC No. 9270 has been revised to address changes made to 10 CFR Part 71 that took effect on October 1, 2004.

Issued with Certificate of Compliance No. 9270, Revision No. 1 on November 24, 2004.