July 6, 2007

Mr. Anthony Patko
Director, Licensing
Engineering
NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, GA 30092

SUBJECT: REVISION 44 OF CERTIFICATE OF COMPLIANCE NO. 9225 FOR THE MODEL NO. NAC-LWT PACKAGE

Dear Mr. Patko:

By letter dated November 17, 2006, as supplemented March 2, May 3, and June 22, 2007, NAC International (NAC) submitted a revised application in accordance with 10 CFR Part 71 for an amendment to Certificate of Compliance No. 9225 for the Model No. NAC-LWT package to incorporate various changes to the authorized contents and the operating procedures. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

Those on the attached list have been registered as users of the package under the general license provisions of 10 CFR 71.17 or 49 CFR 173.471. This 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. Registered users may request, by letter, to remove their names from the Registered Users List.

If you have any questions regarding this certificate, please contact me or Kim Hardin of my staff at (301) 492-3294.

Sincerely,

/RA/

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9225
TAC No. L24043
Enclosures: 1. Certificate of Compliance No. 9225, Rev. No. 44
2. Safety Evaluation Report
3. Registered Users

cc w/encls. 1& 2: R. Boyle, Department of Transportation
J. Shuler, Department of Energy
Registered Users List

Mr. Anthony Patko
Director, Licensing
Engineering
NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, GA 30092

SUBJECT: REVISION 44 OF CERTIFICATE OF COMPLIANCE NO. 9225 FOR THE MODEL NO. NAC-LWT PACKAGE

Dear Mr. Patko:

By letter dated November 17, 2006, as supplemented March 2, May 3, and June 22, 2007, NAC International (NAC) submitted a revised application in accordance with 10 CFR Part 71 for an amendment to Certificate of Compliance No. 9225 for the Model No. NAC-LWT package to incorporate various changes to the authorized contents and the operating procedures. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

Those on the attached list have been registered as users of the package under the general license provisions of 10 CFR 71.17 or 49 CFR 173.471. This 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. Registered users may request, by letter, to remove their names from the Registered Users List.

If you have any questions regarding this certificate, please contact me or Kim Hardin of my staff at (301) 492-3294.

Sincerely,

/RA/

Robert A. Nelson, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9225
TAC No. L24043
Enclosures: 1. Certificate of Compliance No. 9225, Rev. No. 43
2. Safety Evaluation Report
3. Registered Users
cc w/encls. 1 & 2: R. Boyle, Department of Transportation
J. Shuler, Department of Energy
Registered Users

Distribution: Closes L24043
SFPO r/f BWhite RBellamy,RI JMadera,RIII BSpitzberg,RIV DCollins,RII
via e-mail: MRahimi NOsgood SBrown JGlenny
Filename: C:\FileNet\ML071900440.wpd

OFC	SFST	E	SFST		SFST				SFST	
NAME	KHardin		MDebose		ABarto		REinziger		RParkhill	
DATE	06/29/2007		06/29/2007		07/03/2007		07/05/2007		07/02/2007	

OFC	SFST	E	SFST		SFST		SFST		SFST	
NAME	DTang		LCampbell		GBjorkman		MWaters		RNelson	
DATE	06/29/2007		07/06/2007		07/06/2007		07/05/2007		07/06/2007	

C = COVER E = COVER & ENCLOSURE N = NO COPY OFFICIAL RECORD COPY

SAFETY EVALUATION REPORT

Docket No. 71-9225
Model No. NAC-LWT Package
Certificate of Compliance No. 9225
Revision No. 44

TABLE OF CONTENTS

SUMMARY 1

REFERENCES 1

1.0 GENERAL INFORMATION 1

2.0 STRUCTURAL 3

3.0 THERMAL 5

4.0 CONTAINMENT 6

5.0 SHIELDING 6

6.0 CRITICALITY 9

7.0 PACKAGE OPERATIONS 11

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM 11

CONDITIONS 11

CONCLUSION 12

SAFETY EVALUATION REPORT

Docket No. 71-9225
Model No. NAC-LWT Package
Certificate of Compliance No. 9225
Revision No. 44

SUMMARY

By letter dated November 17, 2006, as supplemented March 2, May 3, and June 22, 2007, NAC International (NAC) submitted a revised application in accordance with 10 CFR Part 71 for an amendment to Certificate of Compliance (CoC) No. 9225 for the Model No. NAC-LWT package to incorporate various changes to the authorized contents and the operating procedures. NAC requested inclusion of miscellaneous irradiated hardware and unirradiated fuel for each fuel type as authorized contents; changes to the fuel specifications for TRIGA fuel; clarification of the drain tube and drain alignment ring; and updates to Chapter 7, "Operating procedures," Chapter 8, "Maintenance and Testing Procedures," and various drawings.

EVALUATION

The submittal was evaluated against the regulatory standards in 10 CFR Part 71, including the general standards for all packages, and performance standards under normal conditions of transport (NCT) and hypothetical accident conditions (HAC). Staff reviewed the application using the guidance in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" and NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

Based on the statements and representations in the application, as supplemented, and the conditions listed in the CoC, the staff concludes that the design has been adequately described and evaluated and meets the requirements of 10 CFR Part 71.

REFERENCES

NAC International, application dated November 17, 2006.

NAC International, supplements dated March 2, May 3, and June 22, 2007.

1.0 GENERAL INFORMATION

1.1 Package Description

The Model No. NAC-LWT package with its requested contents is shipped by truck, within an ISO container, or by railcar, as a Type B(U)F-96 package, as defined in 10 CFR 71.4. NAC requested that the package be authorized for the transport of miscellaneous irradiated hardware, unirradiated fuel for each fuel type authorized, and changes to the fuel specifications for the TRIGA fuel.

Other details of the revision request are described below:

This amendment request includes clarification of the use of the drain tube and drain alignment ring as an optional operational feature and deletes references to the obsolete standard drain cover design.

Chapter 8, "Acceptance tests and Maintenance Program," Chapter 7, "Package Operations," and some drawings were updated to incorporate the above changes and provide some clarifications.

Minor editorial changes were made throughout the SAR without changing the technical content.

1.2 Packaging Drawings

One new license drawing is included in this submittal:

Drawing 315-40-145, Rev. 0	Irradiated Hardware, Lid Spacer, LWT Cask

Two updated license drawings were included in this submittal:

Drawing 315-40-02, Rev. 20 (Sheets 1-2)	Body Assembly
Drawing 315-40-08, Rev. 17 (Sheets 1-5)	Cask Parts Details

License drawing 315-40-145 was added to present and define a spacer required for transporting irradiated hardware. The revised drawings were updated to clarify changes for the use of the drain tube and drain alignment ring and provide some clarifications.

1.3 Contents

This package has previously been evaluated and authorized for shipment of TRIGA fuel. This revision authorizes revisions to the TRIGA fuel's characteristics based on revised shielding and criticality analyses.

This revision adds irradiated hardware as authorized content and defines the term as follows:

"...solid, irradiated and contaminated fuel assembly structural or reactor internal component hardware, which may include fissile material, 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."

The authorized fuel description was expanded to include unirradiated fuel for each fuel type with this revision.

2.0 STRUCTURAL

The staff reviewed the application to revise the Model No. NAC-LWT package structural design and evaluation to assess whether the package will remain within the allowable values or criteria for NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 2 (Structural Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" and NUREG-1609 "Standard Review Plan for Transportation Packages for Radioactive Material."

2.1 Structural Review

The proposed amendment adds new Drawing 315-40-145 to define the spacer required to be installed on the bottom of the closure lid during use of the cask to transport irradiated hardware. The spacer ensures that the irradiated hardware cannot reposition itself during transport to be above the lead shielded length of the cask cavity. Two licensing drawings were also updated to remove information of no safety significance.

Section 2.6.12.8 recognizes optional use of the DIDO fuel basket drain alignment ring to be placed between the bottom basket module base plate and the cask bottom forging for providing end support for the basket. Similarly, Section 2.6.12.11 was revised, for the ANSTO fuel basket, for the same optional bottom basket module support configuration.

Table 2.2.0-1 incorporates the updated ANSTO basket weight of 862 lbs for consistency. The resulting package weight of 48,826 lbs for shipping the ANSTO fuel continues to be bounded by the design package weight of 52,000 lbs.

The applicant reevaluated structural performance of the DIDO and the ANSTO fuel baskets subject to the 1-ft and 30-ft end drops. Since the two basket designs are similar and the loaded basket weight of 1,487 lbs for DIDO is bounded by that of 1,620 lbs for ANSTO, the staff agrees with the applicant's assessment that the resulting stress margins for DIDO are bounded by those for the ANSTO, as reviewed below.

Section 2.6.12.11.2 presents the quarter-symmetry finite element model for the ANSTO fuel basket. The fuel tubes and base plate are modeled with ANSYS SOLID45 elements. The chamfered base plate periphery, which interfaces with the transition of the cask inner shell and the bottom forging, was considered the governing boundary condition and modeled with CONTAC52 gap elements. For the 1-ft end-drop inertia load of 15.8 g, the maximum stress intensities, which occur at the interface of the fuel tube and base plate, were determined to be 8.2 ksi and 15.6 ksi for the primary membrane and the primary membrane-plus–bending stresses, respectively. The applicant also noted that the bearing stress at the basket bottom base plate chamfer, which interfaces with the cask body, was calculated to be 10.3 ksi. These values are below the at-temperature allowables.

Section 2.7.7.13.2 considers an inertia load of 60 g to evaluate the ANSTO fuel basket for the 30-ft end drop. The calculated maximum stress intensities are 31.3 ksi and 59.0 ksi, which are below the allowables, for the primary membrane stress and the primary membrane-plus–bending stress intensities, respectively.

On the basis of the review above, the staff concludes that the proposed design changes do not affect the ability of the package to meet the structural performance requirements of 10 CFR Part 71.

2.2 Material Properties

This amendment request includes irradiated hardware and a change to the fuel specification for the TRIGA fuel as authorized contents to the CoC that were included in the materials review.

Unirradiated fuel of each type previously authorized, including TRIGA fuel, was added to the approved contents. There are no compatibility issues associated with these materials due to the fact that the materials are the same, and structural properties of the unirradiated fuel material and cladding such as yield and ductility will be improved. Solid, irradiated, and contaminated fuel assembly structural or reactor component hardware, which may have a limited fissile content, better known as Greater than Class C Waste (GTCC), has been added as an authorized content. The staff reviewed the compatibility of these materials with the canister below.

The applicant listed the hardware materials and stated stainless steel secondary containment would be used when needed to prevent interactions. Since precautions are taken to avoid interaction between two dissimilar metals, the staff agrees that there are no interaction issues and has no materials issues with the addition of these contents to the approved contents.

The term “failed fuel” has been changed to “damaged fuel.” The definition of “damaged fuel” has been modified to include both rod and assembly defects. Any assembly defects will be analyzed for structural soundness. In those cases where the defect impairs structural soundness, the assembly will be considered “damaged” and not be shipped. The staff agrees that the new definition is acceptable.

Based on the information presented in this revision request, staff concludes that there are no new material issues raised. The staff agrees that the changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

2.3 Conclusions

Based on the analyses presented in the NAC-LWT Revision No. 44 SAR submittal, staff concludes that the transport of the contents requested will provide adequate margins of safety during NCT and HAC and is bounded by the previously approved analyses. The staff agrees that the changes do not affect the ability of the package to meet the structural integrity requirements of 10 CFR Part 71.

3.0 THERMAL

The staff reviewed the application to revise the Model No. NAC-LWT package to verify that the package thermal design has been described and evaluated under NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 3 (Thermal Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" and NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

3.1 Thermal Review

This amendment makes numerous changes to the NAC-LWT transportation package as documented in Attachment A to the November 17, 2006, submittal letter. However, the only change made to Chapter 3, "Thermal," was the deletion of reference to the standard port cover which has no effect on the thermal evaluation.

Another change made in other chapters of this amendment that could have an effect on the thermal evaluation included a change of content for the MTR fuel with shortened cooling time based on limited burn-up. However, since the heat load of this package did not change, this will have no impact on the thermal evaluation.

Step 32 of Section 7.1.1, "Procedure for Wet Loading of LWR Fuel Assemblies and Canistered LWR Fuel Rods," states that the cask is filled with helium. However, the thermal model for HACs described in Section 3.5.1.2, refers to Section 3.4.1.7.2 for a detailed description of the thermal model. Section 3.4.1.7.2 identifies air as the gas in the cask cavity. The staff viewed modeling air as the gas in the cask cavity as a non-conservative assumption for the determination of maximum cladding temperature resulting from the HAC fire, since the air has a conductivity of approximately 5 times less than helium and would tend to insulate the fuel from the fire temperatures. In the applicant's response dated June 22, 2007, NAC stated that they changed the model (for Condition 2) to include the conductivity of helium as compared to air and reanalyzed for the maximum cladding temperature. The results from this analysis lowered the maximum calculated cladding temperature from 1014 ^{o}F to 838 ^{o}F. Based on the analyses described in its June 22, 2007 supplement, the applicant agreed to clarify the associated changes in the SAR which will be included in the next amendment request.

As additional assurance that the cladding temperatures aren't exceeded during the regulatory fire, the results from the Baltimore Tunnel Fire (ref. NUREG/CR-6886, Rev. 1), which is a bounding fire to the regulatory fire, illustrates (ref. Figure 7.20) that the peak and average cladding temperature for the NAC-LWT should be below the short term temperature limit of the zirc cladding (1058 ^{o}F).

3.2 Conclusions

Based on the review of the application, the staff found reasonable assurance that the applicant has demonstrated that the evaluation of the NAC-LWT package with the requested changes do not affect the ability of the package to meet the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT

The staff reviewed the application to revise the Model No. NAC-LWT package to verify that the package containment design has been described and evaluated under NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 4 (Containment Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" and NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

4.1 Containment Review

This revision makes numerous changes to the NAC-LWT transportation package as documented in Attachment A to the November 17, 2006, application. However, the only change made to Chapter 4, "Containment," was the deletion of reference to the standard port cover.

Other changes made in other chapters of this amendment that could have an effect on the containment evaluation included a change of content, including irradiated hardware, TRIGA fuel characteristics, and MTR fuel with shortened cooling time based on limited burn-up. This revision also includes the addition of Parker seal compound identification number for the Viton seals used in the alternate and Alternate B port covers, addition of the spacer required to be installed on the bottom of the closure lid for transport of irradiated hardware, and a shortened pre-shipment leakage test duration to 5 minutes for the alternate port covers.

Staff considers all of the requested content changes acceptable because they are bounded by the current design basis fuel, 25 BWR Rods-56% Failed Fuel Fraction, which results in a reference leakage rate of 5.5 x 10E-7 std cm^3 / s (helium) as shown in Table 4.2-4 of the SAR. The change to the identification number of the Vitron seal is acceptable because it does not result any different performance characteristics of the seal. The addition of the spacer on the bottom of the closure lid does not change any of the leakage test requirements of the package, and therefore, is acceptable. Finally, shortening the time duration of the pre-shipment leakage test for the alternate port cover plates to 5 minutes does not compromise the validity of that test. The test continues to demonstrate no leakage for the tested leak rate.

4.2 Conclusions

Based on the statements and representations in the application, staff agrees that the applicant has shown that the requested revision for the Model No. NAC-LWT continues to meet the containment requirements of 10 CFR Part 71.

5.0 SHIELDING

The applicant provided a shielding evaluation for the package containing the requested content changes in Chapter 5 of the application. The staff reviewed the application to revise the Model No. NAC-LWT package to verify that the package shielding design has been described and

evaluated under NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 5 (Shielding Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" and NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

5.1 Shielding Review

The applicant revised the shielding analysis for the Model No. NAC-LWT to consider irradiated non-fuel hardware in the cask cavity. The source term for the irradiated hardware consists primarily of gamma emitters, particularly ^{60}Co, from activated metals. The hardware may contain a low level of gamma and neutron emitting surface contamination, but this material will have no significant effect on external cask dose rates. Additionally, fissile material contamination will be present, but its quantity will be exempt from classification as fissile material for transportation, per the requirements of 10 CFR 71.15.

The applicant determined the source term for the irradiated hardware by assuming one kg of stainless steel with 1.2 g/kg cobalt impurity is activated in a PWR in-core neutron spectrum for the equivalent of a fuel assembly burnup of 45 GWd/MTU, and then cooled for 90 days. The source term for one kg of irradiated stainless steel is then scaled up to a source strength of $2x10^{14}$ γ/s, or $2.2x10^{14}$ MeV/s. The gamma source term for one kg of irradiated stainless steel is shown in the SCALE 18-group structure in Table 5.3-139 of the SAR. The scaled up, limiting source term is shown in the following table:

Table 1: Limiting Gamma Source Term for Irradiated Hardware

Energy Group (MeV)		**Source**	
Upper	**Lower**	**gamma/sec**	**MeV/sec**
1.00E+07	8.00E+06	0.000E+00	0.000E+00
8.00E+06	6.50E+06	0.000E+00	0.000E+00
6.50E+06	5.00E+06	0.000E+00	0.000E+00
5.00E+06	4.00E+06	0.000E+00	0.000E+00
4.00E+06	3.00E+06	2.613E-14	9.080E-14
3.00E+06	2.50E+06	1.187E+05	3.241E+06
2.50E+06	2.00E+06	7.655E+08	1.711E+09
2.00E+06	1.66E+06	9.135E+10	1.659E+11
1.66E+06	1.33E+06	3.226E+13	4.789E+13
1.33E+06	1.00E+06	1.142E+14	1.321E+14
1.00E+06	8.00E+05	3.922E+13	3.505E+13
8.00E+05	6.00E+05	6.222E+09	4.325E+09
6.00E+05	4.00E+05	5.938E+12	2.948E+12
4.00E+05	3.00E+05	4.549E+12	1.581E+12

3.00E+05	2.00E+05	5.576E+10	1.384E+10
2.00E+05	1.00E+05	2.624E+11	3.909E+10
1.00E+05	5.00E+04	6.497E+11	4.838E+10
5.00E+04	1.00E+04	2.744E+12	8.174E+10
	Total	2.000E+14	2.200E+14

The applicant modeled the source as a variable height, variable radius, and zero density volume within the NAC-LWT cavity. Modeling the source with no density conservatively neglects any self-shielding that would be provided by the activated metals. The applicant modeled the NAC-LWT package and the irradiated hardware contents under NCT and HAC, with the only difference between the two models being the lack of the neutron shield under HAC.

The applicant used the SAS2H isotopic depletion and decay sequence of the SCALE code system with the 27-group ENDF/B-IV cross section library to determine the source term for the irradiated hardware. The applicant performed dose rate calculations using the SAS1 one-dimensional, discrete ordinates shielding sequence of the SCALE code system. In the SAS1 model, the applicant varied the height of the source region to produce the highest external radial dose. The axial dose was not determined, since previous shielding analyses for various other contents have demonstrated that radial dose rates are limiting.

The resulting external dose rates are presented in Figures 5.3-76 through 78 of the SAR as a function of source radius and height for NCT at the package surface and two meters from the package surface, as well as for the HAC at one meter from the package surface. Note that the external dose rates for the package, with irradiated hardware contents, exceeds the limits for non-exclusive use shipment in §71.47(a), and therefore must be shipped under the exclusive use provisions of §71.47(b). The maximum external dose rates are summarized for all conditions in the following table:

Table 2: Maximum Dose Rates for NAC-LWT Package with Irradiated Hardware

Dose Location	Normal Conditions of Transport (mrem/hr)	Hypothetical Accident Conditions (mrem/hr)
Surface	300	N/A
One Meter	N/A	61
Two Meters	7.6	N/A

The staff reviewed the applicant's shielding analysis and found it to be conservative, with resulting maximum dose rates within the limits prescribed by shielding requirements of 10 CFR Part 71. Additionally, the dose rates were found to be consistent with or bounded by the previously approved shielding analyses for various other contents. Note that the limiting source term for the irradiated hardware material is less than that for the

majority of other approved contents for the NAC-LWT. The applicant has shown and the staff agrees that the Model No. NAC-LWT continues to meet the external radiation requirements of §71.47 and §71.51(a)(2), for irradiated hardware material with a source term of up to 2.0×10^{14} gammas/sec and 2.2×10^{14} MeV/sec.

5.2 Conclusion

Based upon the information provided by the applicant, the staff has reasonable assurance that the applicant's shielding analyses demonstrate that the package design meets external radiation standards in 10 CFR Part 71.

6.0 CRITICALITY

The applicant provided a criticality evaluation for the package containing the requested content changes in Chapter 6 of the application. The staff reviewed the application to revise the Model No. NAC-LWT package to verify that the package criticality safety design has been described and evaluated under NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 6 (Criticality Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" and NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

6.1 Criticality Safety Review

The applicant revised the criticality safety analysis to demonstrate reactivity results for a modified set of TRIGA fuel element characteristics for non-poisoned baskets. The revised analysis considered a non-poisoned basket containing: 1) 120 intact TRIGA fuel elements, 2) severely damaged TRIGA fuel, including debris, in sealed or screened cans in the top and bottom modules of the package, or 3) TRIGA elements with potential clad damage, but meeting structural requirements for transport, in screened cans in the top and bottom modules of the package. The Model No. NAC-LWT was previously approved for the first two of the above contents.

For the revised TRIGA fuel analysis, the applicant considered both enrichment levels of the fuel, roughly 20 weight percent and 70 weight percent, as well as the maximum fissile material mass for each enrichment/ fuel clad material combination modeled. Analyses summarized in Table 6.4.5-20 of the SAR demonstrate that the 70 weight percent enriched fuel with stainless steel cladding is the most reactive when modeled in a finite cask array. Other parameters which were determined by the applicant to result in the maximum reactivity included: maximum zirconium mass and hydrogen-to-zirconium ratio; maximum rod diameter and fuel expanded to the clad inner surface; reduced clad thickness; maximum outer fuel diameter and minimum inner fuel diameter; minimum central zirconium rod diameter; and nominal active fuel length.

For the intact fuel contents, the applicant modeled four TRIGA fuel elements in each available basket cell in the non-poisoned basket. The applicant varied the rod pitch within the limits of the basket cell opening, the basket cell opening size, the water density in the cask cavity, and the size of the package array. The applicant also

considered the package under NCT and HAC, with the main difference being the absence of the neutron shield under HAC. The maximum reactivity occurred for an array of eight packages under HAC, with a fully moderated cask cavity, and with the fuel elements shifted out to the basket cell corners. The maximum k_{eff} for the revised intact TRIGA fuel description, adjusted for bias and uncertainty, is 0.94842, and the resulting criticality safety index (CSI) is 12.5.

For severely damaged TRIGA fuel elements in screened or sealed cans, the applicant modified the intact fuel cask model to include two elements per can in each basket cell of the top and bottom modules in the cask. The intact fuel was modeled in the most reactive configuration determined from the previous evaluation, with fuel in the screened or sealed cans modeled as either intact or as fuel debris filling the can cavity. Moderator density was varied in the cask cavity as well as preferentially within the cans in order to find the most reactive condition. The applicant found that the system reactivity is dominated by the intact fuel in this situation, with little variation in k_{eff} with varied conditions in the screened or sealed fuel cans. The maximum k_{eff} was calculated to be 0.4971, adjusted for bias and uncertainty, for an array of eight casks under HAC. The resulting CSI for this package configuration is 12.5.

For TRIGA fuel elements with cladding damage but retaining structural integrity for transport, the applicant modified the intact fuel cask model to include four elements per can in each basket cell of the top and bottom modules in the cask. The intact fuel in the central three modules was modeled in the most reactive configuration from the intact fuel evaluation. The canned fuel, with cladding damage as defined in Section 1.2.3.1.1.3 of the SAR, but otherwise intact, was modeled as four intact fuel elements in each screened can in each available basket cell of the top and bottom modules. Moderator density was varied within the cask cavity, as well as preferentially within the screened fuel cans. The applicant found the most reactive condition to be an array of four packages under HAC with full density water in the cavity and 0.4 g/cm^3 density water in the cans. The maximum k_{eff}, adjusted for bias and uncertainty, was 0.94618, and the resulting CSI for this configuration is 25.

The applicant used the SCALE 4.3 code system with KENO V.a and the 27-group ENDF/B-IV cross-section library for all criticality calculations. The benchmarking analysis for TRIGA fuel analyzed with this code and cross-section library is included in Section 6.5.3 of the SAR, and is the same as was approved for earlier TRIGA fuel analyses for this package. The combined bias and uncertainty determined in this analysis, and applied in all the calculations discussed above, is 0.0168.

6.2 Conclusion

The applicant has shown and the staff agrees that the Model No. NAC-LWT package with a non-poisoned basket continues to meet the criticality safety requirements of 10 CFR Part 71 when limited to:

1. 120 intact TRIGA fuel elements,

2. 12 screened fuel cans, each with up to four TRIGA fuel elements with cladding damage as defined in Section 1.2.3.1.1.3 of the SAR, but

otherwise intact, in the top and bottom modules only, with intact fuel in the remaining modules, or

3. 12 screened or sealed fuel cans, each with up to two severely failed TRIGA fuel elements, in the top and bottom modules only, with intact fuel in the remaining modules.

7.0 PACKAGE OPERATIONS

The staff reviewed Chapter 7 of the SAR in the application to revise the Model No. NAC-LWT package to verify that it meets the requirements of 10 CFR Part 71 and is adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

The chapter includes the procedures for package loading, unloading, and preparation of the empty package for transport. To support this revision request, Sections 7.0, 7.1, 7.2, and 7.3 of the SAR were revised to reflect the contents changes, clarify the use of the drain tube and drain alignment ring as an optional operational feature, clarify operational steps, and eliminate redundancy.

Based on the statements and representations in the application, the staff concludes that the package operations meet the requirements of 10 CFR Part 71 and that they are adequate to assure the package will be operated in a manner consistent with its evaluation for approval. Further, the CoC is conditioned to specify that the package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The staff reviewed the revisions to Chapter 8 of the application to verify that the revised acceptance tests for the packaging meet the requirements of 10 CFR Part 71.

To support this revision request, Sections 8.1, 8.1.2, 8.1.3, and Table 8.2-1 of the SAR was revised to enhance and clarify the descriptions and functions of the alternate port cover and alternate B port cover designs and other further clarifying information.

Based on the statements and representations in the application, the staff concludes that the revised acceptance tests for the packaging meet the requirements of 10 CFR Part 71. Further, the CoC is conditioned to specify that each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

CONDITIONS

In addition to the packaging drawings (Condition Nos. 5(a)(3)(i) and 5(a)(3)(ii)) listed in Section 1.2 and the authorized contents listed in Section 1.3 of this SER (Condition Nos. 5(b)(1) and 5(b)(1)(i-iv), (vi-x), (xii), (xiv), and (xvi)), the CoC has been revised as follows:

Condition No. 5.(a)(2):

A reference to 5.(b)(1) was added to replace the entire listing of contents to provide clarity.

Condition No. 5.(b)(2):

The words “or other internal support structure” was added to provide for the spacers needed to support the irradiated hardware.

Condition No. 5.(b)(2)(iii):

Deleted due to the fact that the contents associated with it in 5.(b)(1)(iii) were deleted.

Condition No. 5.(b)(2)(vii):

Revised to reflect the changes in the TRIGA fuel characteristics.

Condition No. 5.(b)(2)(xvii):

Added to reflect the addition of the irradiated hardware.

Condition No. 5.(c)(2) and (4):

Added to reflect the CSIs for the TRIGA fuel.

Condition No. 9.:

Revised to reflect the change in leakage test requirements.

Condition No. 12:

Added the irradiated hardware.

Condition No. 18:

Allows the use of Revision 43 of this certificate for one year.

CONCLUSION

Based on the statements and representations in the application, as supplemented, and the conditions listed above, the staff concludes that the Model No. NAC-LWT package design has been adequately described and evaluated and that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9225, Revision No. 44,
on July 6, 2007.