UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 4, 2014

Mr. Barry K. Miles
Division of Naval Reactors
U.S. Department of Energy
Washington, DC 20585

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9793, REV. NO. 16 FOR THE MODEL NO. M-140 PACKAGE

Dear Mr. Miles:

As requested by your application dated November 5, 2013, as supplemented on July 2, 2014, enclosed is Certificate of Compliance No. 9793, Revision No. 16, for the Model No. M-140 package. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's safety evaluation report is also enclosed.

The U.S. Department of Energy, Division of Naval Reactors, has been registered as a user of the package under the provisions of 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 Bernard White of my staff at (301) 287-0810.

Sincerely,

/RA/

Michele Sampson, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9793
TAC No. L24857

Enclosures:
1. Certificate of Compliance No. 9793, Rev. No. 16
2. Safety Evaluation Report

cc w/encls: R. Boyle, Department of Transportation
J. Shuler, Department of Energy c\o L. T. Gelder

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DATE	10/27/14	10/27/14	10/28/14	10/28/14	10/27/14
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NAME	HLindsay	MDeBose	CAraguas	MRahimi	MSampson
DATE	10/27/14	10/27/14	10/28/14	10/31/14	11/04/14

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SAFETY EVALUATION REPORT
Docket No. 71-9793
Model No. M-140 Package
Certificate of Compliance No. 9793
Revision No. 16

SUMMARY

By application dated November 5, 2013, as supplemented on July 2, 2014, the U.S. Department of Energy, Division of Naval Reactors, requested an amendment to Certificate of Compliance No. 9793 for the Model No. M-140 package. Naval Reactors requested use of an improved grapple adapter design for use with S8G fuel assemblies, reduced thermal and shielding hold times for S8G fuel modules, and revision of the criticality safely index based on calculations for the array after the tests for hypothetical accident conditions.

The U.S. Nuclear Regulatory Commission (NRC) staff performed its review of the M-140 package utilizing the guidance provided in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." Based on the statements and representations in the application, as supplemented, the analyses performed by the applicant demonstrate that the package provides adequate structural, thermal, containment, shielding, and criticality safety protection under normal conditions of transport and hypothetical accident conditions, therefore the NRC staff concludes that the package meets the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

1.1 Packaging

The M-140 is a stainless steel package for transporting spent fuel. The overall cask dimensions are 98 inches in diameter and 194 inches high. The package body is 14 inches thick with a closure head that is secured by 36 wedge assemblies located radially around the inside diameter. Penetrations in the closure head and body include an access port for fuel loading, vent and drain ports, water inlet and outlet penetrations, and a thermocouple penetration. The cask closure head and penetrations are sealed with plugs and double ethylene propylene O-ring seals. A stainless steel protective dome is positioned over the closure head. The cask body has 180 external vertical cooling fins, and a support ring is welded to these cooling fins. The support ring is bolted to a rail car mounting ring during transport. The fuel is positioned within an internals assembly. The internals assembly is composed of stacked spacer plates that have openings for the spent fuel modules. The maximum weight of the package, including contents, is 375,000 pounds.

The applicant requested use of an improved grapple adapter for use with S8G fuel assemblies. Minor modifications to the description of the package were provided along with revised drawings in Section 1.4 of the application.

1.2 Contents

Naval Reactors requested reduced shielding and thermal hold times for S8G fuel modules to 89 and 105 days, respectively, and revised the criticality safety index (CSI) from 100 to 0.

1.3 Conclusion

The changes made to the General Information section were adequate and do not affect the continued ability of the package to meet the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

2.0 STRUCTURAL EVALUATION

The applicant requested modifications to the grapple hardware resulting in component weight and location changes.

The NRC staff reviewed the Appendix 2.10.1 re-analysis of the 30-ft top and bottom package drop hypothetical accident condition tests to account for the heavier weight of the redesigned grapple adapter. For the top drop, the revised structural analysis conservatively ignores the energy dissipation associated with the grapple adapter crushing for an estimate of maximum amount of space for control rod motion. For the bottom drop, the re-analysis also shows that the larger inertia load exerted on the bottom of the fuel module is still acceptable in that the fuel cladding stress remains below the yield strength. By a conservative assumption of complete crushing of the pedestal in the bottom of the container, the re-analysis maximizes the amount of space in the container for calculating the control rod motion, which is subsequently considered for the criticality safety evaluation. On this basis, the staff has reasonable assurance to conclude that the grapple adapter hardware changes were adequately evaluated for meeting the 10 CFR Part 71 requirements.

3.0 THERMAL EVALUATION

The major changes related to the thermal evaluation were to reduce the required hold time of the contents, in terms of days after shutdown. The applicant removed some of the conservatism that was present in the thermal analysis for this package to maintain critical component temperatures within limits given the shortened cooling time.

3.1 Decay Heat

The required hold time for shipment which is given as days after shutdown is determined by calculating the decay heat for contents that takes into account the actual power history of the core. The applicant’s evaluation used a best estimate power history and worst case peaking factors to demonstrate how, using a qualified computer code, shipment times and decay heat rates for contents can be determined. Fuel performance assessment calculations show, and the staff agrees, that fuel blister formation or fuel blister rupture will not occur during normal conditions of transport or hypothetical accident conditions respectively given the required thermal hold time. The applicant provided a decay heat limit that must be reached before shipment can take place. The applicant used a conservative model to determine the allowable decay heat limit for contents, and the staff agrees that the values derived would be bounding for actual shipments.

3.2 Normal Conditions of Transport

The applicant's analysis, which included two different two-dimensional models and two different three-dimensional models, applied the normal conditions of transport described in 10 CFR 71.71 to the package. The staff reviewed the assumptions and methods made by the applicant regarding the thermal performance of the M-140 package. The staff also reviewed the material properties and analysis modeling approach for the normal conditions of transport evaluation, the staff found these to be acceptable. The staff agreed the analysis demonstrated that component temperature limits were met for the limiting decay heat for the specified contents. The package accessible surface temperatures for transport were determined, and the limits for exclusive use shipments in 10 CFR 71.43 were met.

3.3 Hypothetical Accident Conditions

The thermal evaluation for hypothetical accident conditions considered the results of a 30-foot free drop, puncture, and fire tests. The staff also reviewed the material properties and analysis modeling approach for the hypothetical accident conditions evaluation, the staff found these to be acceptable. The model assumed that structural damage from the drop and puncture tests was minimal, and therefore, the same thermal models used for the normal conditions of transport analyses were essentially used for the hypothetical accident conditions analyses. The staff found this approach to be acceptable. The applicant analyzed a fire exposure of 1475°F for 30 minutes, in accordance with 10 CFR 71.73. The staff agreed that an appreciable amount of hot gas from the regulatory fire will not enter the M-140 containment boundary. The staff agreed that the temperature limits prescribed by the applicant are for the performance of the contents, and these limits were not exceeded for the hypothetical accident conditions fire test.

3.4 Conclusions

Based on its review of the methods, analyses, and information presented in the application, the NRC staff agrees with the applicant's conclusion that the thermal requirements of 10 CFR Part 71 will be met with the proposed contents and packaging design.

4.0 CONTAINMENT EVALUATION

The applicant provided a containment evaluation for the S8G spent fuel in the M-140 shipping container. The fuel cladding and other related weldments is what is considered the primary containment boundary for prevention of the release of fuel fission products. Structural and thermal evaluations for the fuel show that cladding integrity is maintained under normal conditions of transport and hypothetical accident conditions. The package containment boundary (also identified by the applicant as the secondary containment boundary) includes the container body (bottom and cylindrical shell), closure head, and closure head access plug. There are also several penetrations closures, including vent, drain, leak test fixture, and water level control plugs. The closure head, the closure head access plug, and other small penetrations are sealed with double O-ring seals.

4.1 Normal Conditions of Transport

The NRC staff reviewed the applicant's containment evaluation for normal conditions of transport and agrees that the M-140 package with S8G spent fuel satisfies the requirements in 10 CFR 71.51 and this analysis is bounded by the containment analysis in the "Core

Independent M-140 Safety Analysis Report for Packaging" (M-140 SARP), which has previously been shown to meet the containment requirements in 10 CFR Part 71.

4.2 Hypothetical Accident Conditions

After evaluating the package for the hypothetical accident conditions tests, the applicant's analysis indicated that the package would not remain leak-tight under all accident conditions. Under certain circumstances, the permanent gap could open between the closure head and container flange as a result of the worst case hypothetical drop accident. The applicant's evaluation has shown that any releases for the limiting contents from the M-140 package do not exceed the release limits in 10 CFR 71.51. In addition, staff agrees that the assumptions used in the M-140 SARP are applicable and conservative for the S8G contents, therefore the proposed changes to the S8G contents in the M-140 container meets the containment requirements in 10 CFR 71.51 limits after the tests for hypothetical accident conditions.

4.3 Conclusions

The NRC staff concludes that, based on the containment evaluation for the S8G spent fuel and the conditions listed in the certificate of compliance, the package meets the containment requirements of 10 CFR 71.51.

5.0 SHIELDING EVALUATION

The applicant proposed reducing the hold time prior to shipment for S8G fuel assemblies and provided a shielding evaluation to support the change.

5.1 Description of Shielding Design

5.1.1 Design Features

The package consists of a cylindrical, lower shell sealed with a flat closure head. The M-140 is normally shipped via rail car, on which it sits inside a well ring and is held by a support ring integral to the car. Items to be shipped in the package include spent fuel modules and assembly internals.

5.1.2 Summary Tables of Maximum Radiation Levels

Maximum Radiation Levels for Normal Conditions of Transport (mrem/hr)

On Contact with Package Surface		Vertical Plane 2 m from Surface of Package	
Calculated	10 CFR 71 Limit	Calculated	10 CFR 71 Limit
105.8	200	9.9	10

Maximum Radiation Levels for Hypothetical Accident Conditions (mrem/hr)

1 Meter from Package Surface	
Calculated	10 CFR 71 limit
118	1,000

Radiation levels are summarized in Tables 5.1-1 and 5.1-2 in the application for normal conditions of transport and hypothetical accident conditions, respectively. The applicant conservatively ignores any shielding provided by the support and well rings when calculating the dose rates 2 meters from the side of the rail car.

5.2 Radiation Source

The applicant conservatively determines source strength by basing it on a bounding power history of any fuel module at any location within the core. Crud is ignored as the source term is negligible compared to the other radiation sources.

5.2.1 Gamma Source

The applicant modeled the gamma source with axial variations, including segments below and above the fuel region. The applicant based the fission product gamma distribution on the maximum power generated by the most depleted fuel assembly at any time in core life. Total gamma strength distributions are provided in Tables 5.2-3 and 5.2-4 of the application.

The applicant conservatively includes the effects of neutron absorption by the fission products in addition to the fission product decay.

The applicant determined activation rates for fast and thermal incident neutrons with a sufficient number of gamma energy groups. The applicant evaluated neutron fluxes in the space above and below the fuel assembly with a 2D discrete ordinates transport code similar to the DOT computer code and based the axial distribution of the source term from those fluxes.

The applicant varied the point in the cycle of a fuel assembly based on the location in which it would be stored in the M-140 package. For side and top radiation levels, the applicant assumed an irradiation history that results in the most axially peaked source strength near the top of the fuel. For bottom dose rates, the applicant chose an irradiation history that produced the greatest source strength at the bottom of the fuel. This approach conservatively results in bounding external dose rates.

5.2.2 Neutron Source

The applicant based the neutron source on the worst-case parameters for any fuel module at any location within the core. The applicant included both photo-neutrons and transuranic neutrons and the effect of subcritical multiplication in determining the neutron source strength. The applicant chose an irradiation history for photo-neutron and transuranic neutron sources that maximized each individually. The applicant determined subcritical multiplication from the most reactive fuel modules in the core. Axial photo-neutron distribution is based on the gamma source.

The applicant used a point depletion code to calculate the transuranic neutron source. This calculation was based on a fuel module with the highest depletion density with an additional conservative factor multiplied to the result.

The applicant chose the ^{244}Cm spontaneous fission energy spectrum to represent photo-neutrons and transuranic neutrons, and chose ^{235}U fission spectrum for subcritical multiplication. The applicant's analysis also accounted for (α,n) reactions. Total neutron source as a function of energy is shown in Table 5.2-5 of the application.

5.3 Shielding Model

The applicant described and/or provided drawings of sufficient detail for NRC staff to confirm the applicant's analysis for both normal conditions of transport and hypothetical accident conditions.

5.3.1 Configuration of Source and Shielding

The applicant conservatively ignores any additional shielding provided by the container support ring, railcar well ring, external cooling fins, top plate, lower supports, grapple adapters, the bottom energy absorber, and the protective dome.

The applicant reduced the thickness of shielding components based on the requirements of 10 CFR 71.51 and determined the most limiting test and location of greatest impact to external dose rates. The applicant displaced the internal components in the model under hypothetical accident conditions in a conservative manner. The applicant's model assumed preferentially flooded components to maximize calculated radiation levels.

5.3.2 Material Properties

The material properties specified in the model are appropriate for the actual construction and contents of the package. A summary of mass and number densities used in the analysis is presented in Tables 5.3-1 and 5.3-2 of the application.

5.4 Shielding Evaluation

5.4.1 Methods

The applicant calculated gamma radiation using a point-kernel code, and neutron radiation using a two-dimensional discrete ordinates code. Both of these methods and the cross-section libraries used have been previously determined to be acceptable.

5.4.3 Flux-to-Dose-Rate Conversion

The applicant used gamma flux-to-dose conversion factors that are contained in the point-kernel code library. The applicant applied the neutron flux-to-dose conversion factors to the fluxes generated by the neutron transport code. The applicant presents these factors in Tables 5.4-1 and 5.4-2 of the application.

5.4.4 External Radiation Levels

The external radiation levels scale directly with the intensity of the source term. The methods and data used to determine the external dose rate are acceptable. The applicant has shown the worst-case source term to be bounding of all expected contents.

5.5 Conclusion

While the applicant's analysis shows that the dose rate limits will be met for an 89-day cool time, since the thermal hold time is limiting, the certificate has been conditioned to require a minimum of 105-day cool time after shutdown. Based on NRC staff review of the methods, analyses, and information presented in the application, the NRC staff confirmed the applicant's conclusion that the shielding requirements of 10 CFR Part 71 will be met with the proposed contents and packaging design.

6.0 CRITICALITY EVALUATION

The applicant provided a revised criticality evaluation for the S8G fuel in the M-140 package as part of the application. The fuel composition and parameters are unchanged from previously approved revisions to the M-140 package. Based on the revised structural analyses, it was determined by the applicant that the control rods could be withdrawn to a greater degree than the original assumption. The applicant also updated their calculations for the CSI to demonstrate that a close-packed infinite array of packages would yield a CSI of 0.

Staff reviewed the proposed change in the maximum withdrawal limit of the control rods and found that this increased the maximum k_{eff} (including all biases and uncertainties) under hypothetical accident conditions, was still considerably below the regulatory limit. Based on NRC staff review of information provided by Naval Reactors, the applicant modeled this increase conservatively and the NRC staff has reasonable assurance of continued subcriticality of the system.

Staff also reviewed the change in CSI from 100 to 0 based on the assumption of an infinite array of close-packed packages would remain subcritical, and based on the discussions with Naval Reactors, agrees that the new CSI should be 0.

Since the resulting k_{eff}s for the evaluated system under both normal conditions of transport and in the worst-case accident configuration were found to be less than 0.95, staff concludes that the Model M-140 containing a full load of S8G fuel modules under the new assumptions continues to meet the criticality safety requirements of 10 CFR Part 71.

CONDITIONS

The certificate of compliance conditions have been revised to incorporate the following changes:

Condition No. 5.(b)(2)(ii) has been revised to state that the decay heat for S8G spent fuel, shall not exceed 47,050 Btu/hr decay heat per package and decay heat limits for prototype spent fuel modules has been removed.

Condition 5.(c) has been revised to change the CSI for S8G fuel from 100 to 0.

Condition 7.(b) has been revised to state that the minimum fuel cooling time is 105 days after shutdown

The References section has been updated to include the supplements submitted by Naval Reactors in the course of the review leading to this amendment.

CONCLUSIONS

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. 9793,
Revision No. 16 on November 4, 2014 .