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

September 26, 2017

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

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9793, REVISION NO. 18, FOR THE MODEL NO. M-140 PACKAGE

Dear Mr. Miles:

As requested by your application dated June 10, 2016, as supplemented on May 1, 2017, enclosed is Certificate of Compliance No. 9793, Revision No. 18, 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) 415-6577.

Sincerely,

/RA/

John McKirgan, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9793
CAC No. L25128

Enclosures:
1. Certificate of Compliance
   No. 9793, Revision No. 18
2. Safety Evaluation Report

cc w/encls: R. Boyle, U.S. Department of Transportation
J. Shuler, U.S. Department of Energy

Issued with CERTIFICATE OF COMPLIANCE NO. 9793, REVISION NO. 18, FOR THE MODEL NO. M-140 PACKAGE DATE: September 26, 2017

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ADAMS Package No.: ML17269A136 Ltr: ML17269A138 Enclosure 1: ML17269A137

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SAFETY EVALUATION REPORT
Docket No. 71-9793
Model No. M-140 Package
Certificate of Compliance No. 9793
Revision No. 18

EVALUATION

Summary

By letter dated June 10, 2016, as supplemented on May 1, 2017, the U.S. Department of Energy, Division of Naval Reactors, (certificate holder) requested an amendment to Certificate of Compliance No. 9793 for the Model No. M-140 package.

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 Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

General Information

1.1 Packaging

The M-140 is a stainless steel package for transporting spent fuel. The overall package 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.

1.2 Contents

The applicant requested a number of changes to the S6W fuel assembly characteristics, including updated material information, fuel performance, and revised decay heat limits.

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 10 CFR Part 71.

2.0 STRUCTURAL EVALUATION

The applicant requested the use of updated cladding material and provided additional control rod insertion and withdraw values.

The NRC staff reviewed updated cladding calculations used by the applicant to demonstrate adequate cladding performance during normal conditions of transport and hypothetical accident conditions. The applicant's calculations showed that positive margins of safety with respect to yielding of the cladding were maintained during normal conditions of transport and hypothetical accident conditions test conditions such as the 30-foot side drop. The applicant also used actual material coupon values to demonstrate that cladding stresses were below yielding for the conditions and tests outlined for hypothetical accident conditions.

The applicant updated control rod insertion and withdraw values based on tolerances and clearances of the control rod and fuel assembly during a 30-foot side drop scenario. The applicant demonstrated that previously determined withdraw and insertion values were still valid.

2.1 Materials Evaluation

The applicant updated material properties for Zirconium alloy, Condition C, utilizing continuous material property equations at various temperatures. In addition, the applicant revised justification for the prevention of Zircaloy brittle fracture.

Although the applicant changed the material properties for Zirconium alloy under Condition C, they used the existing Condition B as their limiting condition for structural calculations where the drawing specifications permit the use of both conditions. The staff reviewed the updated temperature dependent material properties for Condition C and determined that they provide reasonable assurance for safety of the package based on the applicant's use of the most limiting or bounding material properties for structural calculations.

The applicant stated that scoping calculations for fuel regions and control rod welds demonstrate acceptable brittle-fracture margin based on end-of-life hydrogen concentration and applied stress in fuel region which bound the most limiting stress calculations used for hypothetical accident conditions.

The staff notes that Zircaloy and Zirconium-based alloys are materials used in the cladding of fuel and form the primary containment boundary. During operation, corrosion of fuel cladding surfaces generate hydrogen due to the reaction with surrounding water, which diffuses into cladding materials. Hydrogen can cause embrittlement, reducing fracture toughness which could lead to brittle-fracture. The staff reviewed various literature identifying the chemistry of the Zircaloy alloy utilized, which reduces hydrogen absorption significantly and retains excellent corrosion resistance. The staff determined that the Zircaloy provides reasonable assurance for safety of the package based on the applicant's use of the most limiting stress calculations, for hypothetical accident conditions, which are bounded revised scoping calculations.

The staff finds that the M-140 transportation package meet the regulatory requirements of 10 CFR 71 for preventing or mitigating galvanic or chemical reactions, is not adversely affected by cold temperatures and is constructed with materials and processes in accordance with acceptable codes and standards.

2.1.1 Base and Weld Metal Fracture Toughness:

Regulatory Guides 7.11 and 7.12, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches," and "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)," respectively, do not apply to the use of stainless steels, which preclude brittle fracture under both normal conditions of transport and hypothetical accident conditions. The staff finds that by avoiding the use of ferritic steels brittle fracture concerns are precluded. Specifically, primary structural packaging components are fabricated of stainless steel. Since this material does not undergo a ductile-to-brittle transition in the temperature range of interest (down to minus 40 ºC), it is safe from brittle fracture. The staff states that in austenitic stainless steel metal the force required to move dislocations, is not strongly temperature dependent and dislocation movement remains high (i.e., will deform more readily under load before breaking) even at low temperatures and the material remains relatively ductile. Regulatory Guide 7.11 states austenitic stainless steel is not susceptible to brittle facture at temperatures encountered in transport.

2.1.2 Chemical or Galvanic Reactions:

Section 2.4.4 of the safety analysis report for packaging (SARP) demonstrates that M-140 components are fabricated from materials not susceptible to unacceptable chemical or galvanic reactions and are fabricated from corrosion-resistant materials. The staff finds that during normal operation the internals will not be subject to continuous exposure to moisture and that water intrusion is not likely to occur during transport. The number of and galvanic potential between the dissimilar alloys used in fabrication of the M-140 package is low. Therefore, the conditions required to create the possibility for galvanic corrosion is small.

2.2 Conclusion

Based on review of the statements and representations in the application, as supplemented, the staff concludes that the structural design has been adequately described and evaluated and that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 THERMAL

The applicant requested an amendment to the certificate for the M-140 package for the S6W fuel. The major changes related to the thermal evaluation were to update portions of the thermal analysis and limits for fuel performance.

3.1 Decay Heat

The content's decay heat profiles were developed using a computer code which uses the 1994 edition of the American National Standards Institute/American Nuclear Society (ANSI/ANS) 5.1, "Decay Heat Power in Light Water Reactors" and modified to account for local peaking factor and additional decay heat from decay of activated core structural material and transuranic actinides. The decay heat data was included in the updated thermal models as heat generation within the fuel zones. The results for the limiting decay heat spent fuel module at each time after shutdown were shown in the application. The decay heat for the high performance fleet core (HPFC) shipboard bounds the other S6W shipboard cores and the advanced fleet reactor (AFR) prototype core.

3.2 Material Properties and Component Specification

Normal transport conditions are limited by the onset of blister formation of spent fuel, and hypothetical accident conditions are limited by blister rupture. Equations were provided for the time to blister formation and the time to blister rupture for normal and accident conditions, respectively. Equations were also provided for spent fuel usage for each interval and total usage. The thermal heat capacity and thermal conductivity of the cladding were also updated.

The spent fuel total usage is evaluated for an infinite dry time which exceeds the 1 year shipment time. Acceptable fuel thermal performance is shown based on the consideration of the fuel usage for the transfer from the reactor vessel to the M-140, the fuel usage for an M-140 drain time which is calculated for a table of values of days after shutdown, and additional dry periods that occur after shipment with margin. The AFR prototype core and S6W shipboard core each has acceptable fuel thermal performance at (or greater than) a specific drain time days after shutdown, and the corresponding bounding decay heat limits for the M-140 with the AFR prototype core and S6W shipboard core were provided in the certificate of compliance. Once a reactor shuts down, the power history of the reactor is used to determine a hold time based on the decay heat limit.

3.3 Normal Conditions of Transport

The applicant's analysis, which included a three-dimensional, 1/8-symmetry model used for evaluations of fuel element performance relative to the limits for blister formation. The applicant also described the spent fuel thermal limits, boundary conditions, and application of decay heat, as well as the qualification and verification of the analysis code. The normal conditions of transport thermal evaluation demonstrated that the spent fuel can reside in the M-140 package indefinitely and that the thermal usage for the shipment will not cause blister formation. The model used to provide temperatures for the structural evaluations has not been changed in this amendment. The staff reviewed the analysis modeling approach for the normal conditions of transport thermal evaluation and finds that the thermal analysis demonstrates that component temperature limits are met for the limiting decay heat for the specified contents. The package accessible surface temperatures for transport were determined in previous analysis, and the temperatures are below the limit for exclusive use shipments in 10 CFR Part 71.43.

3.4 Hypothetical Accident Conditions

The thermal evaluation for hypothetical accident condition considered the results of a 30-foot free drop, and puncture tests. The damage conditions have negligible effect on the heat transfer from the fuel, therefore, the normal thermal model geometry of S6W spent fuel within the M-140 was analyzed for the fire test except it is assumed to have separated from the conveyance. The applicant analyzed a fire exposure of 1475°F for 30 minutes, in accordance with 10 CFR Part 71.73. The staff reviewed the analysis modeling approach for the hypothetical accident conditions thermal evaluation and the temperature limits prescribed by the applicant for the performance of the contents, and finds that the limits were not exceeded for the hypothetical accident conditions fire test. The applicant calculated the fuel usage due to the hypothetical accident conditions fire for the S6W fuel and AFR prototype fuel, and in addition the usage of the completion of the M-140 shipment. The application demonstrated that that there is sufficient spent fuel performance to support the draining of the M-140 at a specific drain time for each of the S6W fuel and AFR prototype fuel, which corresponds to the total decay heat limits in the certificate of compliance.

3.5 Conclusions

Based on its review of the methods, analyses, and information presented in the application, the 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 objective of the review was to verify that the containment associated with the M-140 shipping container transporting S6W spent fuel was adequately described and evaluated under normal conditions of transport and hypothetical accident conditions to meet regulations, as required by 10 CFR Part 71. Previous submittals dealt with spent fuel content with greater potential for release from the M-140 shipping container, such as due to larger quantities of crud. Although there were no significantly new items related to containment, the application included some changes to the containment summary to reflect the bounding nature of the previous submittals with spent fuel content. Regulations applicable to the containment review include 10 CFR 71.31, 10 CFR 71.33, 10 CFR 71.35, 10 CFR 71.43, and 10 CFR 71.51.

According to the containment chapter of the SARP, the containment boundary consists of cladding surrounding the fuel (primary containment) and the containment provided by the M-140 shipping container (secondary containment). The cladding provides containment of the fission products related to the spent fuel; the M-140 shipping container provides containment for the spent fuel as well as contaminated metal and crud exterior to the spent fuel. The applicant noted that the primary containment is verified to not have any cracks during fabrication. In addition, the applicant noted that M-140 shipping container is fabricated from thick steel joined by full penetration welds. The nondestructive testing of the welds is performed in accordance with the requirements of Article NB-5000 (Category A) of the America Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III.

4.1 Normal Conditions of Transport

The containment chapter of the SARP indicated that the primary and secondary containment boundaries are not affected by the normal conditions of transport tests defined in 10 CFR Part 71.71.

4.2 Hypothetical Accident Conditions

The containment chapter of the SARP indicated that fission products are completely contained within the fuel cladding during the tests for hypothetical accident conditions and that the fuel cladding of the S6W spent fuel shipping assembly remains intact. The SARP indicated that previous analyses showed that the M-140 container closure (secondary containment seal) is not maintained following hypothetical accident conditions, although the head remains attached to the body. The SARP indicated that previous analyses of the M-140 shipping worst case spent fuel showed that the post-accident crud release from the secondary containment would be less than an A_2 quantity and would meet 10 CFR Part 71 requirements. The application indicated that because the design elements of the S6W spent fuel, such as less crud and a lower pressure shipping environment, the present content is bounded by the previous analyses, and therefore, the M140/S6W post-accident releases would also meet 10 CFR Part 71 requirements.

4.3 Conclusions

Based on review of the statements and representations in the application, the staff concludes that the containment design of the S6W spent fuel content and the M140 shipping container has been adequately described and evaluated and that the package design meets the containment requirements of 10 CFR Part 71.

5.0 SHIELDING EVALUATION

The applicant requested an amendment to the certificate for the M-140 package. The major change related to the shielding evaluation was the revised S6W spent fuel characteristics. The objective of this review is to verify that the M-140 shipping container meets the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

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 allowed under 10 CFR Part 71 on contact with the package surface and at 2 m under normal conditions of transport are 200 mrem/hr and 10 mrem/hr, respectively. Maximum radiation levels on contact with the package surface under hypothetical accident conditions are 1000 mrem/hr. Radiation levels are summarized in Tables 5-1A and 5-1B, and Tables 5-2A and 5-2B in the SARP for normal conditions of transport and hypothetical accident conditions, respectively, and are below the limits listed above.

5.2 Radiation Source

The applicant indicated that the shielding analysis conservatively determined the source strength by basing it on a bounding core power history. The applicant ignores crud as previous reviews have shown its contribution to the source term is negligible compared to the other radiation sources. The staff reviewed the radiation source and finds this acceptable as it results in the most limiting radiation source.

5.2.1 Gamma Source

The applicant stated it 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 Table 5-4 of the SARP.

Staff observed that the applicant’s model includes assumptions that overestimate the contribution of subcritical multiplication to the gamma source. When determining the gamma source from component activation, the applicant applied administrative factors to the thermal

neutron reaction rate, modeled materials with the maximum amount of activation impurities, and assumed a neutron fall-off rate that maximized component activation. The applicant includes gamma rays induced by the absorption of neutrons by the fission products in addition to those directly from fission product decay. Staff finds the applicant's assumptions acceptable since, according to guidance in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" and NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages" and prior staff review, they have been shown to overestimate the source term and the corresponding external dose rate.

5.2.2 Neutron Source

The applicant based the neutron source on the most highly depleted 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 independent irradiation histories for photo-neutron and transuranic neutron sources that maximized each individually. The applicant assumed a system k_{eff} that overestimates subcritical multiplication. While the applicant includes the source neutrons from control rods, the applicant removes the control rods from the neutron shielding model. Staff finds the applicant's assumptions acceptable since prior staff review has determined they increase the modeled neutron source term and minimize mitigating neutron absorbers, thus maximizing the calculated neutron 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. Staff has found this methodology acceptable in previous review. Total neutron source as a function of energy is shown in Table 5-5 of the SARP.

5.3 Shielding Model

The staff found 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 the presence of the top container dome when calculating expected dose rates at the top of the package under normal conditions of transport. The applicant also conservatively ignores any shielding provided by rail car under hypothetical accident conditions.

The applicant 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 staff determined this assumption is conservative as it takes no credit for the shielding effects provided by material that may be present and provide additional shielding.

The applicant reduced the thickness of shielding components based on the results of the hypothetical accident conditions tests to determine compliance with 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 conservatively combines all of the features of bottom, side, and corner drops and puncture accidents.

Considering the applicant assumed conditions that are generally accepted to be conservative and applied additional administrative safety factors, staff finds the applicant's shielding model acceptable.

5.3.2 Material Properties

Staff reviewed the material properties specified in the model and found them to be 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 5-6 and 5-7 of the SARP.

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. The staff determined these methods and the cross-section libraries used have been previously determined to be acceptable.

5.4.2 Input and Output Data

The applicant did not provide sample input.

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 in addition to a conservative quality factor. Staff finds this acceptable as the conversion factors have been previously reviewed by staff and the conservative quality factor overestimates dose contribution. The applicant presents these factors in Tables 5-8 and 5-9 of the SARP.

5.4.4 External Radiation Levels

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

5.5 Conclusion

Based on staff review of the methods, analyses, information presented in the application, and prior staff review, for the reasons discussed above, staff finds reasonable assurance 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 requested an amendment to the certificate for the M-140 package, with the major change related to the criticality evaluation was the revised S6W spent fuel characteristics. The objective of this review is to verify that the M-140 shipping container continues to meet the criticality safety requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

6.1 Description of Criticality Design

The package is designed to transport spent fuel modules and prevent in-leakage of water under normal conditions of transport. The package internals maintain spacing between the spent fuel modules to reduce neutron interaction between modules, and the thickness of the M-140 package reduces neutron communication between packages in arrays.

6.2 Fissile Material Contents

The M-140 will contain three types of S6W type fuel assemblies, which are analyzed using the most reactive time in life in core, including conservative reactivity allowances. Based on staff review of the parametric studies performed for the various assembly types, NRC staff confirmed that the applicant used the most reactive S6W fuel type with the most reactive core in their calculations.

6.3 General Considerations for Criticality Evaluations

The applicant based the construction of the S6W spent fuel module to represent an accurate geometry model, with special importance given to aspects of the model that are important to criticality safety. Material data used in the evaluation include American Society of Mechanical Engineers, ASTM International, and Naval Nuclear Propulsion Program documents. Calculations are performed using a Monte Carlo neutron transport theory computer program, and the model accurately represent the features of the fuel module in complete detail, including fuel fillers, poison fillers, cladding, coolant channels control rod channels, and the structural material of the module.

The applicant demonstrated the maximum reactivity of the M-140 package by evaluating various sensitivities and assumptions for the single package evaluation under flooding conditions, a single package under normal conditions of transport, a single package under hypothetical accident conditions, and packages in arrays under both normal conditions of transport and hypothetical accident conditions. Based on the application, the criticality safety index (CSI) was calculated for a close packed infinite array of loaded M-140 packages would yield a CSI of 0 in accordance with 10 CFR 71.59(a)(1) and 10 CFR 71.59(a)(2).

6.4 Single Package Evaluation

For the single package evaluation, staff confirmed that the applicant adhered to the applicable conditions of 10 CFR 71.55 and evaluated various flooded conditions as well as residual water left in the package. In all instances, the resulting calculated k_{eff} that was identified by the applicant was found to be less than 0.95, including all biases and uncertainties for both normal conditions of transport and hypothetical accident conditions.

6.5 Evaluation of Package Arrays Under Normal Conditions of Transport

For the array of packages under normal conditions of transport, a close packed hexagonal array was utilized to minimize spacing in an infinite array, with the applicant modeling the M-140 with varying degrees of moderation in compliance with the applicable portions of 10 CFR 71.59. The applicant included a demonstration of determining maximum reactivity utilizing numerous parametric studies for all of the scenarios evaluated. In all instances, the resulting calculated k_{eff} that was identified by the applicant was found to be less than 0.95, including all biases and uncertainties.

6.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

For the array of packages under hypothetical accident conditions, a close packed hexagonal array was again utilized to minimize spacing in an infinite array, with the applicant modeling the M-140 with varying degrees of moderation in compliance with the applicable portions of 10 CFR 71.59. The applicant included a demonstration of determining maximum reactivity utilizing numerous parametric studies for all of the scenarios evaluated. In all instances, the resulting calculated k_{eff} that was identified by the applicant was found to be less than 0.95, including all biases and uncertainties.

6.7 Benchmark Evaluations

The applicant provided an extensive benchmark evaluation that compared calculational methods with experimental results to determine appropriate bias and uncertainties.

6.8 Conclusion

The criticality safety method employed by the applicant complies with the requirements of 10 CFR 71.31(a)(2) and 10 CFR 71.35. Since the resulting k_{eff}s for the evaluated system under both normal conditions of transport and hypothetical accident conditions were confirmed through the applicant's analysis to be less than 0.95, staff concludes that the Model M-140 containing a full load of S6W fuel modules under the assumptions utilized by the applicant continues to meet the criticality safety requirements in 10 CFR Part 71.

CONDITIONS

The following changes have been made to the Certificate:

Item 5.b(2)(iv) was revised to update the maximum decay heat for the shipboard and prototype cores.

Item 9(b) was revised to specify that the minimum time for container draining is specified in the SARP

The “REFERENCES” section was revised to include the date of the letter requesting the amendment and its supplement.

CONCLUSION

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. 18,
on 9/26/17.