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

May 31, 2018

Mr. Roy G. Pratt
NAVSEA 08G
1240 Isaac Hull Ave SE Stop 8021
Washington Navy Yard, DC 20376-8021

SUBJECT: REVISION NO. 22 OF CERTIFICATE OF COMPLIANCE NO. 6386 FOR THE MODEL NO. 235R001 TRANSPORTATION PACKAGE

Dear Mr. Pratt:

As requested by Naval Reactor's application dated June 9, 2017, enclosed is Certificate of Compliance (CoC) No. 6386, Revision No. 22, for the Model No. 235R001 transportation package. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The U.S. Nuclear Regulatory Commission staff's safety evaluation report is also enclosed.

If you have any questions regarding this certificate, please contact 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. 71-6386
EPID No. L-2017-LLA-0062

Enclosures:
1. CoC No. 6386, Rev. No. 22
2. Safety Evaluation Report

REVISION NO. 22 OF CERTIFICATE OF COMPLIANCE NO. 6386 FOR THE MODEL NO. 235R001 TRANSPORTATION PACKAGE, DOCUMENT DATE: May 31, 2018

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ADAMS Package No.: ML18152A982 Letter & SER.: ML18152A983
CoC: ML18152A984

OFC:	DSFM	DSFM	DSFM	DSFM
NAME:	BWhite	SFigueroa Via email	SEverard via email	JSmith via email
DATE:	4/23/18	5/7/18	4/24/18	5/4/18
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SAFETY EVALUATION REPORT
Docket No. 71-6386
Model No. 235R001
Certificate of Compliance No. 6386
Revision No. 22

EVALUATION

By application dated June 9, 2017, the U.S. Department of Energy, Naval Reactors, requested and amendment to Certificate of Compliance (CoC) No. 6386, for the Model No. 235R001 transportation package. The applicant requested deletion of the S3G fuel refueling cells as authorized contents and changes to various criticality safety indexes (CSIs). The NRC has reviewed the amendment and finds that the changes do not affect the ability of the package to meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

The NRC staff performed its review of the amendment to the 235R001 certificate of compliance utilizing the guidance provided in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

Structural Evaluation

The Naval Reactors Program uses the 235R001 shipping containers to ship A1G fresh fuel cells to servicing facilities. During the development of the A1G configured M-290 spent fuel transportation package safety analysis report (A1G SAR), the applicant determined that the A1G lead screw and tie-rod were susceptible to brittle fracture failure in the M-290 shipping container as a result of the hypothetical accident conditions in 10 CFR 71.73. In addition, the applicant stated that this failure mode had not been considered for the A1G fresh fuel assembly in the 235R001 shipping container SAR (A1G 25-181 SAR). Specifically, failure of the lead screw invalidates the control rod position assumption used in previous criticality analyses.

In addition to the lead screw analysis, the applicant also stated that the A1G fuel cells are planned to transition to a lower enrichment. As a result, the applicant performed an analysis of the lower enriched A1G fuel assembly in the 235R001 shipping container against the requirements of 10 CFR 71.

The scope of the structural safety evaluation is limited to reviewing the applicant's analysis of the lead screw and their assumptions for the position of the control rod used in the criticality analysis under both normal conditions of transport and hypothetical accident conditions, as well as evaluating the analysis of the lower enriched A1G fresh fuel transported in the 235R001 shipping container.

2.1 Description of Structural Design

With the exception of the enrichment of the fuel cells, neither the structural design of either the Naval Reactors 235R001 shipping container, nor the A1G contents were changed via this amendment.

2.2 Normal Conditions of Transport

In Appendix (1) of Enclosure (12) of the A1G 25-181 SAR, the applicant compared the higher and lower enriched A1G fuel cells with respect to materials, dimensions and weight to determine any differences that would warrant a new analysis. The applicant compared the drawings of both fuel cell types and determined that the specified materials, specified dimensions, and weights were the same for both fuel types (with the exception of the enrichment of the fuel material). The applicant stated that the weights of the fuel clusters often have a weight variability due to construction tolerances, which includes different fuel loadings. Although the offset between the fuel loadings may result in a slightly different weight for the lower enriched fuel, it is well within the specified tolerance, which are considered in all models and analyses.

As a result of their analysis, the applicant determined that the structural calculations for the package of the higher enriched fuel bound those of the lower enriched fuel and that shipment of the lower enriched fuel was covered by the existing structural analysis.

The staff reviewed the applicant's analysis and determined that all structural analyses conducted on the higher enriched fuel, to include finite element models, bound all applicable analyses for the lower enriched fuel and that the structural assumptions for normal and accident condition package configurations for criticality evaluations remain unchanged. Because of this, the staff finds that the 235R001 transportation package, loaded with the lower enriched A1G fuel cells, continues to satisfy the applicable structural requirements of 10 CFR 71.

In Appendix (2) of Enclosure (12) to the A1G 25-181 SAR, the applicant determined the load capacity of the lead screw with respect to brittle fracture and compared this value to the loads encountered during normal conditions of transport. To do this, the applicant first determined the stress required to produce a brittle fracture in the lead screw material. This level of stress occurs when the brittle fracture stress intensity factor equals the fracture toughness of the material. The applicant then used this stress value to determine the inertial load required to achieve this level of stress in the material. The applicant then compared this load capacity with the resultant inertial loads produced by the various tests proscribed for normal conditions of transport in 10 CFR 71.71. The applicant stated that the inertial load required to produce a brittle fracture in the lead screw material was greater than two times the load produced by normal conditions of transport (vibration, shock, and the 3-foot flat bottom drop tests).

The applicant also noted that four decades of acceptable 235R001 shipments of A1G fresh fuel cells provides supplementary evidence that the A1G cells will not be affected by normal conditions of transport loads.

The staff reviewed the applicant's calculations for determining the inertial load required to produce a brittle fracture in the lead screw material. Because the load required to produce a brittle fracture is more than twice that produced by the proscribed normal conditions of transport tests, the staff finds that there is no substantial reduction in the effectiveness of the packaging that would prevent it from satisfying the requirements of 10 CFR 71.55(d)(2) for a fissile material package.

2.3 Hypothetical Accident Conditions

Because the applicant conceded lead screw failure during the hypothetical accident conditions tests in 10 CFR 71.73, it did not provide further structural calculations on the A1G fuel assembly and assumed complete control rod withdrawal for the subsequent criticality evaluation.

Based on the applicant's statement in the SAR, the staff finds that the geometric form of the package contents is substantially altered under the hypothetical accident conditions tests in 10 CFR 71.73, in that failure of the lead screw due to brittle fracture could cause complete withdrawal of the control rod which will subsequently increase the reactivity of the package. Further evaluation of the package content with respect to satisfying the subcriticality requirements of 10 CFR 71.55(e) for a fissile material can be found in section 6 of this SER.

2.2 Findings

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

Criticality Evaluation

6.0 Criticality Safety Evaluation

The 235R001 shipping container is a transportation package that is designed to ship fresh A1G fuel cells. In addition, the applicant also requests authorization for transport of the A1G fuel cells at a lower enrichment than previously approved.

During the NRC review of the M-290 transportation package, the applicant identified that the A1G lead screw was susceptible to brittle fracture failure in the M-290 transportation package under hypothetical accident conditions, and recognized that this failure mode was not analyzed for criticality safety of the 235R001 shipping container containing the A1G fuel, which could potentially affect the control rod position and, therefore, the reactivity of the contents within the package. The scope of the criticality safety review is limited to the applicant's revised criticality safety analysis of the package resulting in altered control rod position due to failure of the lead screw, and evaluation of the proposed lower enrichment of the A1G fresh fuel to be transported in the 235R001 shipping container.

6.1 Description of Criticality Design

The criticality design of the 235R001 shipping container remains unchanged from the previously approved analysis. However, the identified lead screw failure during hypothetical accident conditions results in an unanalyzed condition for the potential control rod position in the A1G fuel cells. The applicant addressed this unanalyzed condition by assuming the controls rods are removed from the modeled fuel cells, taking no credit for neutron absorption effects of the control rods.

6.2 Fissile Material Contents

The Naval Reactors program is transitioning enrichment of the A1G fuel cells to a lower enrichment than was previously approved. This nominal decrease in enrichment results in a reduction of the fissile mass contents in the A1G fuel cell, and consequently, a decrease in the neutron multiplication factor in comparison to the original fuel design.

6.3 General Considerations for Criticality Evaluations

The applicant's analysis is based on the drawings of the A1G fresh fuel package 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 the American Society of Mechanical Engineers, ASTM International, and Naval Nuclear Propulsion Program documents. Calculations are performed using a Monte Carlo neutron transport theory-based computer program, and the model accurately represents the features of the fuel cells in complete detail, including fuel, poison, cladding, coolant channels, control rods and channels, and the structural material of the A1G fuel.

The applicant demonstrated the maximum reactivity of the 235R001 shipping package loaded with any A1G fuel cell by evaluating various sensitivities and assumptions for the single package evaluation under flooded 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 CSI calculated for a close packed infinite array of loaded 235R001 packages is 100 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, an infinite close packed hexagonal array was utilized to minimize spacing in an infinite array with the applicant modeling the 235R001 shipping container loaded with any type of A1G fuel cell with varying degrees of moderation consistent with the applicable portions of 10 CFR 71.59. The applicant included a demonstration of maximum reactivity utilizing numerous parametric studies for all of the scenarios evaluated, including the normal location of the control rods. In all instances, the calculated k_{eff} was found by the applicant 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, the applicant modeled a single 235R001 shipping container loaded with any type of A1G fuel cell with varying degrees of moderation consistent with the applicable portions of 10 CFR 71.59. The applicant included a demonstration of maximum reactivity utilizing numerous parametric studies for all of the scenarios evaluated. With respect to the potential for lead screw failure resulting in control rod movement, the control rods were assumed to be fully withdrawn from the A1G fuel cells. Therefore, no control rod portions were modeled, which increases the reactivity due to less neutron absorption and is a conservative assumption. In all instances of the studies, the resulting calculated k_{eff} were found by the applicant to be less than 0.95, including all biases and uncertainties. The removal of the control rods from the fuel cells results in an increase in the CSI from 25 to 100, which limits transport of the loaded 235R001 shipping container to one package per shipment.

6.7 Benchmark Evaluations

The applicant provided an extensive benchmark evaluation that compared calculational methods with experimental results to determine appropriate bias and uncertainties. NRC staff reviewed the benchmarking evaluation methodology, discussed the methodology with Naval personnel at previous site visits to Bettis Atomic Power Laboratory, and the staff determined that the methodology employed complies with the requirements of 10 CFR Part 71.

6.8 Evaluation Finding

As a result of the structural review indicating that failure of the lead screw could result in the complete withdrawal of the control rods, the NRC staff reviewed the applicant's analysis with regards to the lead screw failure evaluation and the resultant increase in reactivity in criticality analysis. NRC staff also reviewed the applicant's analysis of a reduction in fuel enrichment to ensure that the applicant's analysis adequately bounded the 235R001 package with this lower enriched A1G fuel. Since the resulting k_{eff} for the evaluated system under both normal conditions of transport and hypothetical accident conditions were confirmed by the NRC staff review of the applicant's analysis to be less than 0.95, staff concludes that the 235R001 shipping container containing any type of A1G fuel cell under the assumptions utilized by the applicant, and under the conditions listed in the certificate of compliance, continues to meet the criticality safety requirements in 10 CFR Part 71.

CONDITIONS

The package description in 5.(a)(2) was revised to remove contents that were removed as authorized contents.

Content in Condition No. 5(b)(1)(iv) for the S3G fuel has been deleted, content Condition Nos. 5(b)(1)(v), 5(b)(1)(vi), 5(b)(1)(vii), and 5(b)(1)(viii) were renumbered throughout.

The CSI for Content No. 5.(b)(1)(i) was changed to 100.

The CSI for Content Nos. 5.(b)(1)(iv), and 5.(b)(1)(v) were reserved for future use.

The references section was updated to include this request.

CONCLUSION

Based on the statements contained in the application, and the conditions listed above, the staff concludes that the changes indicated do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 6386, Revision No. 22,
on 5/31/18.