71-9023

NAC INTERNATIONAL

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February 28, 1997

Mr. Eric J. Leeds, Section Chief
Spent Fuel Licensing Section
Spent Fuel Projects Office
Office of Nuclear Material Safety and Safeguards
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, Maryland 20555-0001

Dear Mr. Leeds:

Reference: 1. USNRC Certificate of Compliance (COC) No. 71-9023, Revision 7, Model No. NLI-10/24 Package.

NAC International (NAC) hereby requests the timely renewal of the Certificate of Compliance No. 71-9023 for the Model No. NLI-10/24 Packaging (Package Identification No. USA/9023/B( )F) in accordance with the provisions of 10 CFR 71.38.

In conjunction with this request , NAC has reviewed the NLI 10/24 Safety Analysis Report (SAR) to ensure that it is complete and current, specifically the sections on operating procedures, acceptance tests, and maintenance programs. No needed changes in these sections were identified. In the review of the other SAR sections, it was identified that the text references to Fissile Class III needed to be replaced by the proper Transport Index terminology per Reference 1 and 10 CFR 71.59. Three sets of the revised pages are provided with this submittal for your use.

If you have any questions or comments regarding the enclosed information, please contact me at (770) 447-1144. Thank you for your consideration of this renewal request..

Sincerely,

T. C. Thompson

Thomas C. Thompson
Manager, Licensing
Engineering & Design Services

NTOI 1/

cc: C. J. Haughney (USNRC)

ED970106

NLI 10/24

Safety Analysis Report

February, 1997

NAC International
655 Engineering Drive
Norcross, GA 30092

Decay heat is removed form the fuel to the cask by thermal radiation and conduction through an aluminum fuel basket. Heat is then transferred through the cask sides and ends by a combination of conduction, natural convection in the water filled neutron shield, and natural convection and radiation from the surfaces of the cask. Being entirely passive this means of heat dissipation is highly reliable. A self-contained redundant auxiliary cooling system, mounted on the rail car, is used to maintain cask and fuel temperatures as low as possible solely to facilitate cask cooldown and unloading operations.

The criticality analysis was based on fresh fuel assemblies with zero burnup and shows that the NLI 10/24 rail cask meets all the requirements of 10 CFR 71.59 for a package with a Transport Index of 100.

The NLI 10/24 cask has been designed to meet all applicable requirements of 10 CFR, Chapter 1, Part 71 and 49 CFR, Chapter 1, Parts 170-189. The design function was carried out under N L Industries, Inc. Quality Assurance Program for Design and engineering which is comparable to 10 CFR, Chapter 1, Part 50, Appendix B and ASME, Section III, Design Control. Manufacturing and Quality Assurance Program will be carried out in accordance with N L Industries, Inc. Commercial Nuclear Quality Control Manual as applicable by design requirements.

4.0 Criticality Analysis

The NLI 10/24 package is assigned a Transport Index for criticality control of 100 in accordance with 10 CFR 71.59. The fresh fuel assumption was used in performing the required analyses. A preliminary analysis was used to determine the most reactive pitch of the individual fuel assemblies assuming water filled cavity conditions. This most reactive pitch was used in performing the criticality analysis for the array. Results are summarized in Table II-2. A detailed description of the analysis is provided in Section X.

TABLE II-2
CRITICALITY ANALYSIS RESULTS

Fuel Type	Number of Elements	Weight Percent U-235 in Fuel	K_{eff}
PWR	10	3.50	0.946 ± 0.005
BWR	24	2.80	0.953 ± 0.005

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.22 (a) (4)	Identification and volumes of any coolants and of receptacles containing coolant;	Being a dry shipment with passive heat removal, there is no coolant associated with the package
71.22 (b) (1)	Identification and maximum radioactivity of radioactive constituents;	See Section III, "Fuel Description and Source Data."
71.22 (b) (2)	Identification and maximum quantities of fissile constituents;	Section III, "Fuel Description and Source Data."
71.22 (b) (3)	Chemical and physical form;	See Section III, "Fuel Description and Source Data."
71.22 (b) (4)	Extent of reflection, the amount and identity of non-fissile neutron absorbers;	See Section X, "Criticality Analysis".
71.22 (b) (5)	Maximum weight of contents	The maximum weight of fuel carried in the cask is 18,000 pounds.
71.22 (b) (6)	Maximum amount of Decay Heat;	The maximum decay heat generated by fuel assemblies in the cask will not exceed 70kw.
71.23	Deleted	

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.24	Procedural Controls	Procedural Control adequate to satisfy the requirements of 71.51 (b) have been prepared. See 71.51 below.
SUBPART C		
71.31	General Standards for all Packaging	
71.31 (a)	Packaging shall be of such materials and construction that there will be no significant chemical, galvanic, or other reaction among the packaging components.	There will be no significant chemical, galvanic, or other reaction among the packaging components, or between the packaging components and the package contents. The fuel is carried in an aluminum basket which is housed in the stainless steel weldment. Reaction between local uranium shielding and stainless steel is prevented by a layer of flame-sprayed copper applied to those areas of the stainless steel shell where maximum temperatures are predicted
71.31 (b)	Packaging shall be equipped with a positive closure which will prevent inadvertent opening.	The double containment system utilized two separate closure heads. The inner head is held in place by 16 stud bolts, 1 ¾" diameter. The outer closure head is held in place by 28 bolts, 1¼ inch diameter. Removal of the closure head requires deliberate action and the use of tools.

Section of Part 71	Requirement or Subject of Provision	Assessment of Compliance
71.36 (b)	Criticality	The cask would remain sub-critical following the hypothetical accident sequence specified in Appendix "B" of this part. See Section X, "Criticality Analysis".
71.37	Evaluation of Fissile Package Array	
71.37 (a)	Model Testing	Model testing has not been done to evaluate the cask by the criteria specified in 71.39. Damage to the package, following the hypothetical accident described in Appendix "B" of this part, has been evaluated by the analytical methods. See response to 71.34 (a) (2) above.
71.37 (b)	Criticality Assumptions	The assumptions made in determining compliance with 71.39 (a) (2) comply with this subpart. See Section X, "Criticality Analysis".
71.38	Deleted	
71.39	Deleted	
71.40	Deleted	
71.41	Previously Constructed Package	Not Applicable.