July 29, 2010

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE CERTIFICATE OF COMPLIANCE NO. 9235, REVISION FOR THE MODEL NO. NAC-STC PACKAGE TO INCORPORATE THE DAIRYLAND COOPERATIVE' S LACROSSE BWR FUEL AS AUTHORIZED CONTENTS

Dear Mr. Patko:

By letter dated December 17, 2009, as supplemented April 28 and June 3, 2010, NAC International requested the U.S. Nuclear Regulatory Commission to revise the Model No. NAC-STC package Certificate of Compliance No. 9235, to incorporate the Dairyland Power Cooperative's LaCrosse Boiling Water Reactor (LACBWR) as authorized content.

In connection with our review, we need the information identified in the enclosure to this letter. Additional information requested by this letter should be submitted in the form of revised Safety Analysis Report pages. To assist us in scheduling staff review of your response, we request that you provide this information by August 20, 2010. If you are unable to provide a response by that date, our review may be delayed.

Please reference Docket No. 71-9235 and TAC No. L24408 in future correspondence related to this request. The staff is available to meet to discuss your proposed responses. If you have any questions regarding this matter, I may be contacted at (301) 492-3339.

Sincerely,

/RA/
Kimberly J. Hardin, Senior Project Manager
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9235
TAC No. L24408

Enclosure: Request for Additional Information

July 29, 2010

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Director, Licensing
Engineering
NAC International
3930 East Jones Bridge Road, Suite 200
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Enclosure: Request for Additional Information
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NAME	KHardin		MDeBose		JPiotter		REinziger		VWilson	
DATE	7/23/10		7/23/10		7/27/10		7/28/10		7/27/10	
OFC	SFST	E	SFST		SFST		SFST		SFST	
NAME	RParkhill		DJackson		CRegan		MRahimi		EBenner	
DATE	7/26/10		NA		7/28/10		7/29/10		7/29/10	

C=Without attachment/enclosure E=With attachment/enclosure N=No copy OFFICIAL RECORD COPY

Request for Additional Information
NAC International
Docket No. 71-9235
Certificate of Compliance No. 9235
Model No. NAC-STC Package

By letter dated December 17, 2009, as supplemented April 28 and June 3, 2010, NAC International requested the U.S. Nuclear Regulatory Commission revise the Model No. NAC-STC package Certificate of Compliance (CoC) No. 9235, to incorporate the Dairyland Power Cooperative's LaCrosse Boiling Water Reactor (LACBWR) fuel as authorized contents.

This request for additional information (RAI) identifies information needed by the U.S. Nuclear Regulatory Commission staff in connection with its review of the application. The requested information is listed by chapter number and title in the applicant's Safety Analysis Report (SAR). NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," was used by the staff in its review of the application.

Each individual RAI describes information needed by the staff for it to complete its review of the application and to determine whether the applicant has demonstrated compliance with regulatory requirements.

Chapter 2.0 Structural Review

2-1 Justify the use of a 100HZ low-pass Butterworth filter to determine the global response to external impacts.

Staff requested, in the first round of RAIs, justification for the use of 20 g and 48 g/55 g design accelerations for the 1-foot Normal Conditions of Transport (NCT) drop and the 30-foot Hypothetical Accident Conditions (HAC) drop, respectively, since they were not explained in the supplemental SAR sections submitted with this licensing action, nor was any citation provided by the applicant noting their location in the complete SAR. The applicant responded by directing staff to Table(s) 2.6.7.4.2-3 and 2.6.7.4.2-4 of the complete SAR. These tables illustrate the design basis decelerations used to qualify the package as the applicant indicated.

Upon review of these tables and the corresponding SAR section (2.6.7.4.2.6), staff noted that the applicant made use of a low-pass Butterworth filter which used a frequency cutoff of 100 HZ to filter the deceleration time histories for the drop events. This cutoff frequency appears too low based on cutoff frequencies typically used and staff requests a justification for the choice of the cutoff frequency given that a) the low cutoff frequency lowers peak declerations, thereby reducing the effective demand on the structure, which may be erroneous, and b) the margins of safety calculated for components of the structure are very small (+0.05 for example) and are based on a design acceleration that has the potential to be artificially too low.

This information is necessary to determine compliance with 10 CFR 71.71 and 71.73.

2-2 Based on staff's review of round 1 RAI 2-9 response , provide the following:

1- Include in the statement in SAR Section 7.1.3.2, and by reference in the CoC, that, prior to loading a previously-stored canister into a transport cask, that an evaluation will be done to ensure all the requirements of the NAC-STC are met including additional information on what the evaluation will consist of and how it will be performed.
2- Describe how the corrosion evaluation in the marine atmosphere will be conducted and over what period of time it is valid.
3- Define a "loading-specific evaluation" mentioned in the RAI 2-9 response.
4- Provide the basis for the statement, including any documentation, that Duke Power has not damaged any canisters when they removed them from the concrete overpacks.
5- Provide an justification that the neutron absorbers (poisons) in the fuel basket will still function as required after the storage duration, accounting for, at a minimum, boron depletion, thermal and neutron embrittlement, and potential blistering.

The applicant provided a response to the first round RAI question 2-9 addressing the potential aging of a previously stored Multi-Purpose Canister (MPC) and its contents. The response was a statement in SAR Section 7.1.3.2 that "Canisters that are retrieved from storage will be evaluated to ensure that site-specific ambient conditions and potential canister-specific exposure to 10 CFR 72 off-normal and accident conditions meet the design and licensing requirements for a canister to be transported in the NAC-STC package."

The staff finds that the conclusion in the RAI response that code materials licensed for 40 years in-reactor use will survive the rigors of normal or off-normal storage conditions, and as such, these MPC materials will have essentially the same properties after a maximum 40 year storage period. No conclusion is made for periods beyond 40 years. The staff concluded that the material properties may or may not remain the same after an accident and thus does not support the analysis for accident conditions. Because neutron absorbers are non-code materials, the staff does not support the conclusion for these materials and an analysis of the aging effects in these materials must be provided.

The staff agrees that since the fuel burnup will be <45 GWd/MTU and be stored in an inert dry configuration that the condition of the fuel is not expected to change. The staff based their conclusion on the conditions of ISG-11, Rev. 3, being met, and the results of the examination of a cask containing low burnup fuel in Idaho after approximately 15 years storage.

The applicant relies on operating experience at Duke Power to support the conclusion that the MPCs can be removed from the storage overpack with no damage without supporting justification.

This information is needed to meet the requirements of 10 CFR 71.85.

Chapter 5.0 Shielding Review

5-1 Provide additional information on the fuel specifications.

Previous RAI 5-2 requested additional information on the fuel description. The staff requires the additional following information:

(a) Correct the typographical error on the new Table 1.4-4. The 6th row for "Min./Max. Initial Enrichment" is the same as the last row "Maximum enrichment" and appears to have an incorrect footnote. The minimum enrichment for both fuel types (3.6%) is not listed in this table. The staff thinks that the "Min./Max." row was intended for "Minimum enrichment." The staff notes that minimum enrichment is also incorrect in the draft CoC.

(b) Expand the definition of the fuel in the draft CoC. The staff finds that there is not enough information to adequately describe the fuel assemblies. The CoC should include items such as lattice type (array), number of fuel rods and instrument tubes, maximum pitch, minimum rod diameter, minimum cladding thickness, maximum pellet diameter, minimum guide tube thickness, etc. The staff would like for the CoC contents description to be consistent with the NAC-UMS transport canister.

This information is needed to satisfy the requirements of 10 CFR 71.33(b) which gives specific requirements for specifying the contents of the package.

5-2 Provide a definition for "angular lead slump."

Previous RAI 5-4 requested additional information on the lead slump. New Section 5.6.3.3 defines a value for the angular lead slump. The staff does not know what this is. The structural evaluation is defined in terms of radial lead slump.

This information is needed to satisfy the requirements of 10 CFR 71.33(b) which gives specific requirements for specifying the contents of the package.

5-3 Describe the locations of the dose points with respect to the trunnions and heat fins.

Previous RAI 5-6 requested additional information on the locations of the dose points. The response stated that the locations were at the cask centerline. The staff does not understand where the dose points are located azimuthally and needs confirmation that they are conservatively located.

This information is needed to determine that the limits in 10 CFR 71.47 are met.

5-4 Provide a narrative description and/or a cartoon drawing of the neutron shield fins.

Previous RAI 5-9 requested additional information on certain design parameters. The RAI response identified the drawing number illustrating the neutron shield fins as 423-802. The drawing is not clear enough for the staff to understand the neutron shield fins.

This information is needed to satisfy the requirements of 10 CFR 71.33 which states that "the application must include a description of the proposed package."