REQUEST FOR ADDITIONAL INFORMATION
NAC International, Inc.
DOCKET NO. 72-1025
Related to MPC-LACBWR STORAGE SYSTEM
CERTIFICATE OF COMPLIANCE NO. 1025
AMENDMENT NO. 6

By letter dated January 16, 2009 (ML090270151), as supplemented February 11, 2009 (ML090490720), April 1, 2009 (ML090960575), and April 30, 2009 (ML091250187), NAC International, Inc. (NAC) submitted an application to amend Certificate of Compliance No. 1025 and license Technical Specifications in accordance with 10 CFR Part 72. This amendment proposes to modify the NAC-MPC storage system to incorporate Dairyland Power Cooperative (DPC) La Crosse Boiling Water Reactor (LACBWR) spent fuel assemblies as approved contents. The DPC storage system is designated MPC-LACBWR.

This request for additional information (RAI) identifies additional information needed by the U.S. Nuclear Regulatory Commission (NRC) staff in connection with its review of the amendment. The requested information is listed by section. Where applicable, NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" was used by the staff in the review of the amendment application. Each individual RAI section describes information needed by the staff to complete its review of the application and the Safety Analysis Report (SAR) to determine whether the applicant has demonstrated compliance with the regulatory requirements.

STRUCTURAL

3.0 Structural Design

RAI 3.1 Section 3.1.1 Discussion. Revise the SAR text by providing also a description on LACBWR to establish links to Appendix 3.A, as appropriate, to mirror the evaluations performed on CY-MPC and Yankee-MPC.

The subject description is incomplete. It is required for compliance with 10 CFR 72.244.

3.A Structural Evaluation for the MPC-LACBWR Storage System

RAI 3.2 3.A.4.4.1.1 MPC-LACBWR Canister Analysis. Verify that the finite analysis model depicted in SAR Figure 3.A.4.4.1.1 is used for the canister structural evaluation.

The element discretization scheme at the closure lid-to-shell interface region is seen markedly different from that displayed in Figure 11.A.2.12-2. It's unclear whether a relevant finite element model and its analysis are properly reported in the SAR for the canister structural analysis. This information is required for compliance with 10 CFR 72.236(b).

RAI 3.3 3.A.4.4.1.1 MPC-LACBWR Canister Analysis. Clarify SAR-Figure 3.A.4.4.1-2, Detail A, by providing appropriate section annotations indicating the stress reporting summary for the upper left corner for the inner ring.

Enclosure

Details are lacking for a sufficient description of the modeling approach to evaluating stress performance of the canister closure redundant sealing configuration. This information is required for compliance with 10 CFR 72.236 (b).

11.A Accident Analysis for the MPC-LACBWR Storage System

RAI 3.4 <u>11. A.2.12.2.1 Analysis of the MPC-LACBWR Concrete Cask</u>. Clarify the basis for selecting 10,000 psi as the bounding modulus of elasticity in the cask tip-over analysis for the LACBWR ISFSI site. Clarify the basis for the soil depth of 6 ft for the site-specific application.

It's unclear how the soil modulus of elasticity is selected for the ISFSI site, which could be far way from the reactor site and render the soil boring data at the reactor premises inapplicable to the tip-over analysis. This information is required for compliance with 10 CFR 72.236 (b).

RAI 3.5 <u>11. A.3.1 Basket Support Disk Evaluation for the 30-ft Side Drop</u>. Remove the subject evaluation from the SAR.

The side drop evaluation for a 10 CFR 71.73(c) requirement is not relevant to the amendment that is on request. This is required for compliance with 10 CFR 72.244.

RAI 3.6 <u>11. A.3.2 Basket Support Disk Evaluation for the 30-ft Top End Drop</u>. Remove the subject evaluation from the SAR.

The top end drop evaluation for a 10 CFR 71.73(c) requirement is not relevant to the amendment that is on request. This is required for compliance with 10 CFR 72.244.

THERMAL

RAI 4.1 Correct the maximum heat load per assembly of the Yankee-MPC from the stated value of 0.102 kW in Section 4.0, 2nd paragraph, to reflect the value derived from 36 assemblies with a total canister heat load of 12.5 kW and as shown in Table 3.1 “Yankee-MPC Design Basis Fuel Characteristics”.

This information is required for compliance with 10 CFR 72.236 (f).

RAI 4.2 Evaluate and explain in the SAR the thermal effect of compaction of all 32 damaged fuel can's fuel to 50% of there original length. Since damaged fuel has no quantified structural integrity, its ultimate failure and reconfiguration must be presumed and evaluated.

This information is required for compliance with 10 CFR 72.236 (f).

RAI 4.3 Explain why the calculated maximum temperature for the neutron shield of 347F exceeds it temperature limit of 300F in Table 4.A.3-5 “MPC-LACBWR Maximum Component Temperatures for the Transfer Condition- Helium and Vacuum in

Canister". Also, correct the SAR to show a neutron shield maximum temperature of less than the temperature limit of 300F.

Since the heat load of the LACBWR is approximately a factor of 4 less than the CY-MPC, it is not clear why the neutron shield temperature would be exceeded for the LACBWR and not for the CY-MPC as shown in Tables 4.5.3-5 & -6 for transfer operations.

This information is required for compliance with 10 CFR 72.236 (f).

RAI 4.4 Justify the maximum internal pressure of 20 psig used in Section 11.A.2.1.2 "MPC-LACBWR Canister Maximum Stress Due to Internal Pressure".

When using the ideal gas law and the methodology from Section 4.A.3.5 "Maximum MRC-LACBWR Internal Pressure for Normal Conditions" to determine the maximum canister internal pressure under accident conditions, a value of 30 psig is obtained rather than the utilized value of 20 psig. The increase is due to 100% of the spent fuel tubes failing with 30% of the fission gas escaping into the canister and the accident temperature increasing to 680F.

Pressure= (256.3 moles)(0.0821 atm-liter/mole-K)(633K)/ (4371 liters) = 3.04 atm or 30 psig.

This value of 30 psig also exceeds the internal pressure loading used in the stress calculations in Chapter 3, as discussed in the second paragraph of this section.

This information is required to demonstrate compliance with 10 CFR 72.236 (l).

SHIELDING

RAI 5.1 Provide additional information describing the loading pattern used to perform the shielding analyses.

Justify the fuel loading pattern chosen for the shielding analysis is conservative and bounds all possible loading patterns. The staff notes from Figure B2-3 of the technical specifications that there are multiple possible loading patterns.

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

RAI 5.2 Provide additional information about the use of continuous energy cross sections with grouped energy source term.

You use a source term with grouped energy structure for photons and neutrons and input this into a code that has continuous energy cross sections. Did you run MCNP with grouped cross sections? If not, provide information explaining what impact (if any) there is on calculated dose using a discrete energy source term propagated through the cask using continuous energy cross sections.

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

RAI 5.3 Provide additional information regarding the use of homogenized fuel source term when calculating neutron dose.

The staff understands that the fuel source term is homogenized. This may be non-conservative when calculating the neutron dose rate for the flooded condition because homogenized fuel and moderator can show an artificially increase in resonance absorption in U-238. How did you account for this in your neutron dose rate calculations for the flooded condition?

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

RAI 5.4 Provide additional information on the confirmation of boron content in NS-4-FR.

Provide additional information discussing how you determine that the NS-4-FR material has the appropriate boron content and uniformity. Discuss any testing and acceptance criteria.

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

RAI 5.5 Provide additional information on the code used to perform the shielding calculations.

Provide the specific version of MCNP used to perform the shielding calculations as well as the cross section set used.

This information is needed to ensure that the requirements in 10 CFR 72.236 (d) are met.

RAI 5.6 [Editorial] The units for the Fuel Rod Pitch and Displacement Rod Diameter in Table 2.A.1-1 are stated to be centimeters, however the dimensions given appear to be in inches.

CRITICALITY

RAI 6.1 Provide additional information on the assumptions used to calculate the values of k_{eff} on Page 6.A.1-2

Provide the assumptions used in the criticality models that were used to calculate the values of k_{eff} on Page 6.A.1-2. Justify that these conditions are conservative. Include specifically information such as:

a. Loading pattern, including damaged fuel assemblies;

b. Provide additional and specific details of the “combined tolerance model”;

c. Reflector assumptions (outside of the cask);

d. Clad gap flooding; and

e. Details on the flooding (i.e. which areas of the cask/assemblies were flooded with what density water).

The staff understands that sensitivity studies were performed to find the most reactive conditions for these parameters, however the conditions used to determine the k_{eff} values on Page 6.A.1-2 are not explicitly listed.

The staff needs this information to determine that k_{eff} has been calculated with the maximum reactivity and to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

RAI 6.2 Provide additional information on the analyses performed for the damaged fuel.

Provide additional information regarding the assumptions used when modeling a damaged fuel assembly and justify that they are conservative relative to an actual damaged fuel assembly. The staff understands that evaluations were performed to find the maximum reactivity, however the conditions used to determine the k-eff values on Page 6.A.1-2 are not explicitly stated. Address the following concerns related to damaged fuel evaluations:

a. What were the results of the analyses for the unclad rods? For the models that produce the results in the Table on Page 6.A.1-2, state if the damaged fuel assemblies have cladding.

b. The studies for missing rod geometries go up to the removal of 20 rods. Are there any fuel assemblies with more than 20 missing fuel rods? If so, justify that any fuel assemblies with more than 20 missing fuel rods are bounded by the analysis. In addition, justify that the pattern in which the rods were removed is conservative.

c. Provide additional information on the fuel/water mixture studies. Justify the conditions used to simulate fuel rubble are conservative or realistic. Was the density of the mixture varied with height? What was the maximum density used? Justify that it is conservative or realistic.

d. Clarify what is meant by “DFC fuel material is increased by 5% over that of an undamaged assembly” (Section 6.A.3.2.1(e)). What was increased? Density? Active fuel length?

The staff needs this information to determine that k_{eff} has been calculated with the maximum reactivity and to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

RAI 6.3 Provide additional information on off-normal and accident conditions.

There is no discussion of the consideration of the effect of accident conditions on the criticality analyses. Provide additional information demonstrating that the criticality analysis bounds any and all off-normal and accident conditions.

The staff needs this information to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

RAI 6.4 Provide additional information regarding the composition of Boral.

The composition of Boral in Table 6.A.3-9 is inconsistent with that used in the MCNP input deck as well as that used for the other NAC-MPC applications. Provide the specifications of the Boral and justify what was used in the criticality analyses is conservative.

The staff needs this information to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

RAI 6.5 Provide additional information about the “component shift” study.

The staff is not clear about the results of the “component shift” study. Section 6.A.4.4.1, page 6.A.4-5 states that moving the fuel assembly and tube towards the basket center increases system reactivity. However Figure 6.A.3-2 indicates that the criticality analysis was performed with the assemblies shifted away from the basket center as compared to drawing 630045-878. Clarify the results of the “component shift” study and justify that the criticality analysis is conservative.

The staff needs this information to determine that k_{eff} has been calculated with the maximum reactivity and to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

RAI 6.6 Provide additional information on the effect of partial flooding.

This includes scenarios where the cask is partially filled with water, or part water and part steam, or preferential or uneven flooding. The staff recognizes that partial flooding was considered based on a statement in Section 6.A.1 on page 6.A.1-2 that states that “The full moderator density TSC interior condition bounds any off-normal or accident condition with the exception of the preferentially flooded DFC case.” However the staff does not have enough information about the studies performed on partial flooding to understand what was done and that the maximum reactivity was appropriately determined. Clarify the statement in Section 6.A.3.1.1 on Page 6.A.3-2 that states “Partial flooding evaluations are limited to cases with a void region above the active fuel region and canister cavity moderator from the top of the active fuel region to the canister bottom.” Provide additional information describing what was done. Provide figures as necessary illustrating which portions of the fuel/basket were flooded and with what density water. Demonstrate that the criticality analysis is conservative considering the effect of partial flooding.

The staff needs this information to determine that k_{eff} has been calculated with the maximum reactivity and to ensure that the applicant has met the requirement in 10 CFR 72.124(a) and 72.236(c).

RAI 6.7 Provide additional information about the cross sections used.

State which cross section set was used in the criticality analysis. Verify that these cross sections were also used in determination of the USL, or provide justification that the cross sections would provide similar or conservative results.

The staff needs this information to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

RAI 6.8 Provide additional information about the critical benchmark experiments.

Provide a list of the experiments used in the critical benchmark cases.

The staff needs this information to ensure that the applicant has met the requirements in 10 CFR 72.124(a) and 72.236(c).

CONFINEMENT

RAI 7.1 Correct the acceptance criteria for the pressure test as stated in Section 8.A.1.1 “Loading and Closing the TSC”, Step 44.e and as stated in Section 9.A.2.1 “Pressure Testing of the TSC” to be in accordance with ASME Code.

The two aforementioned sections state the acceptance criteria to be “no loss in pressure or visible water leakage from the closure lid weld during the 10 minute test period.” However, the Code requires that the examination for leakage be done after the 10 minute pressure test duration and then the examination is done while maintaining pressure in the TSC equal to the greater of the design pressure or 3/4ths of the test pressure. In no case is holding the test pressure at a constant value for ten minutes considered as a substitute for examination of leakage.

10 CFR 72.236 (l) requires that the spent fuel storage cask must be evaluated, by the appropriate tests or by other means acceptable to the NRC, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal and credible accident conditions.

MATERIALS

RAI 8.1 The acceptance and sampling plan for the neutron absorber in the SAR should be incorporated into the CoC by reference. A statement to the effect that the wet chemistry analysis will be calibrated against neutron absorption methods should be added to the acceptance plan.

Accurate knowledge of the boron content of the absorbers is necessary to assure that the criticality determinations are correct.

This information is required for compliance with 10 CFR 72.236(c).

RAI 8.2 Provide assurance that the 30% fission gas release (FGR) used in the pressurization calculations is bounding for the La Crosse fuel.

SAR (Section 11.A.2.1.1) uses a 30% FGR to estimate the internal transportable storage container (TSC) pressure. This is a good bounding gas release for modern day fuel. Some earlier vintages of BWR fuel exhibited significantly higher fission gas release.

This information is required for compliance with 10 CFR 72.236(d)

RAI 8.3 Specify the minor changes that were made to the vertical concrete cask (VCC)?

The request letter said that minor changes were made to the VCC but never specified the changes. An evaluation of the importance or significance of these changes can not be made without knowing their nature.

This information is required for compliance with 10 CFR Part 72.

RAI 8.4 Provide mechanical and thermal properties of the alloy 348H stainless steel used in the cladding.

The SAR only specifies that cladding as stainless steel. No specific alloy is given although the staff was able to determine it is 348H.

This information is required for compliance with 10 CFR 72.236(a), and 10 CFR 72.236(e).

RAI 8.5 Provide justification or preferably measurement data supporting a maximum fuel assembly bow of 0.25 inches for the Exxon Nuclear Company (ENC) stainless steel clad fuel assemblies.

SAR Section 11.A.4 indicates that a maximum assembly bow of 0.25 inches is expected for the ENC assemblies. While this is reasonable for Zircaloy cladding, it is not clear that it is the maximum for assemblies containing stainless steel clad fuel. Excessive bow would create problems inserts and removing assemblies from the basket tubes.

This information is required for compliance with 10 CFR 72.236(h)

RAI 8.6 Define a “handling container” and specify its properties.

The definition of FUEL DEBRIS (CoC Section A 1.1) says that fuel debris will be placed in a handling container but never defines the term or gives specifications for this container.

This information is required for compliance with 10 CFR 72.236 (a).

RAI 8.7 Define a "fuel bundle."

Item 5 of the LACBWR DAMAGED FUEL ASSEMBLY definition contains the term fuel bundle (CoC Sec A1.1).

This information is required for compliance with 10 CFR 72.236 (a).

RAI 8.8 Revise the definition of "Structural Damage".

The current definition leaves ambiguity in the meaning of intact fuel. Fuel assemblies with structural damage CANNOT be classified as "intact" as indicated in the current definition in Section A 1.1 of the CoC.

This information is required for compliance with 10 CFR 72.236(a).

RAI 8.9 Add the dye penetrant testing (PT) weld inspection criterion for the entire load bearing welds to the drawing of the damaged fuel can. Specify the size of the mesh that will be used at the ends of the damaged fuel cans.

This information is required for compliance with 10 CFR 72.236(j).

RAI 8.10 Clarify the phrase "stainless steel covered BORAL" (SAR Section 1.A.2.1.1). Is the BORAL completely isolated from the pool water when loading the TSC? Evaluate the potential to generate hydrogen by the interaction of any exposed aluminum in the basket, such as the BORAL and heat transfer disks, with the borated pool water during loading. What steps are taken to mitigate the effects of any generated hydrogen?

Any exposed aluminum surfaces in the TSC will react with the borated pool water when the fuel is being loaded. The result of the interaction will be the generation of hydrogen. If sufficient steps are not taken after the lid is put on the TSC, there may be a hydrogen buildup that can possibly be explosive during the welding process.

This information is required for compliance with 10 CFR 72.236(b).

RADIATION PROTECTION

RAI 11.1 Clarify the information in Table 10.A.3-1.

Provide additional clarifying information on the Exposure calculated in Table 10.A.3-1. The values in Step 1 appear inconsistent with the values calculated in Steps 7 and 8. Clarify if the Exposure is listed per person or for total of all personnel and correct the discrepancy.

This information is needed to ensure that the regulations in 10 CFR 20.1201 are met.