July 15, 2005

Mr. William Bracey
Project Engineer
Transnuclear, Inc.
Four Skyline Drive
Hawthorne, NY 10532

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR AMENDMENT NO. 1 TO THE TN-68 STORAGE CASK (TAC NO. L23802)

Dear Mr. Bracey:

By letter dated January 14, 2005, Transnuclear, Inc. (TN) submitted an application for Amendment No. 1 to Certificate of Compliance (CoC) No. 1027 for the TN-68 Storage Cask System. The amendment requested, among several changes, to increase the allowable fuel burnup, minimum cooling times, decay heat, and fuel enrichment. The amendment also requests to include damaged fuel as authorized contents of the cask and to reduce the cask spacing on the storage pad.

The staff has determined that additional information is required to assess compliance with 10 CFR Part 72. Enclosed is the staff's request for additional information (RAI) for the continued review of your request. To the extent practicable, we request that TN respond to this RAI by providing a response to each item in the RAI. We would be willing to meet with you to discuss and clarify the enclosed RAI. Your response to the enclosed RAI is expected by September 16, 2005. If you are unable to meet the September 2005 milestone, you must notify us in writing, at least two weeks prior to this date of your new response date and the reasons for the delay. The staff will assess the impact of the new response date and issue a revised schedule.

Please reference Docket No. 72-1027 and TAC No. L23802 in future correspondence related to this request. If you have questions concerning this request, please contact me at 301-415-1309.

Sincerely,

/RA/

Jose R. Cuadrado, Project Engineer
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 72-1027
TAC No. L23802

Enclosure: Request for Additional Information

Mr. William Bracey
Project Engineer
Transnuclear, Inc.
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Docket No. 72-1027
TAC No. L23802 **ML052000321**

Enclosure: Request for Additional Information

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Request for Additional Information

Docket No. 72-1027
Certificate of Compliance No. 1027
TN-68 Spent Fuel Dry Storage Cask

By application dated January 14, 2005, Transnuclear, Inc. (TN) requested an amendment to Certificate of Compliance No. 1027 for the TN-68 Spent Fuel Dry Storage Cask. This request identifies additional information needed by the U.S. Nuclear Regulatory Commission (NRC) staff in connection with its review of the application. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," was used by the staff in its review of the application. This request describes information needed by the staff to complete its review of the application and to determine whether the applicant has demonstrated compliance with the requirements of 10 CFR Part 72. The requested information is listed by chapter number and title in NUREG-1536.

Chapter 1.0 - General Information

1-1 Revise the applicable drawings for storage cask to contain the following information:

B. The manufacturer's specification (including series, model number, or type) of each coating or sealant to be used, and the component to which these coatings will be applied to on the cask.

10 CFR 72.24(c)(3) requires that information relative to the materials of construction be described in sufficient detail.

Chapter 3.0 - Structural Evaluation

3-1 Provide analyses demonstrating that fuel rod cladding integrity is maintained for the drop scenarios evaluated in SAR Section Appendix 6A, Section 6A.2, "Tipover or Side Drop," Subsection 6A.2.a, "Fuel Rods Supported by Spacer Grids," and Subsection 6A2.b, "Fuel Rod Overhanging at Basket Top End." The analysis assumptions must be justified based on the physical and behavioral characteristics of the fuel rods in the assemblies. *Cladding material properties must be consistent with high burnup fuel and include a thickness reduction due to oxidation*.

<u>Subsection 6A.2.a</u>
The applicant states, "The stresses for different General Electric fuel assemblies are computed in Table 6A-1. It is seen that the 49,422 psi is the highest stress and occurs in GE12 -10x10 fuel assembly." The bending stress component of this total stress is 37,903 psi. In calculating this bending stress, the applicant used a moment of inertia (MI) equal to the sum of the MI's of the fuel tube (0.00050) in^4 and the fuel pellet (0.00070) in^4. This implies that the spent fuel is a solid continuum capable of resisting the same stress as the cladding. Given the fractured state of the spent fuel in storage, the staff finds such an assumption to be inconsistent with the physical state of the fuel, and therefore unacceptable. Using the cladding tube MI to resist bending, the staff calculates bending stress of 91,760 psi.

Subsection 6A.2.b

The applicant's calculations assume that the assembly horizontal inertia forces above the top of the basket are resisted equally by all 78 fuel rods as cantilever beams. This is true only if the grid spacer is located at the top of the fuel basket and bears directly on the top edge of the basket. During an April 11, 2005, teleconference with the applicant, it was revealed that the first grid spacer is located 16.25 inches below the top of the basket. As the grid spacer is located 16.25 inches below the top of the basket, the outside row of ten (10) fuel rods bear directly on top of the fuel basket and will resist the brunt of the horizontal inertia forces. Therefore cladding integrity must be evaluated based on the actual geometry and expected behavior of the assembly.

Because of the non-conservative assumptions made by the applicant, the staff finds the approach in 6A.2.a, and 6A.2.b to be inconsistent with actual fuel rod behavior during a side drop event.

This information is needed to satisfy the requirements of 10 CFR 72.236(b), (c), (d), (h) and (i).

3-2 Provide an analysis demonstrating that fuel cladding integrity is maintained for the end drop event evaluated in SAR Appendix 6A, Section 6A.3, "Bottom End Drop." Analysis assumptions must be justified based on the physical and behavioral characteristics of the fuel rods (cladding and fuel) in the assemblies. *Cladding material properties must be consistent with high burnup fuel and include a thickness reduction due to oxidation*.

The applicant performed an elastic-plastic stress analysis using ANSYS Finite Element Program where the inertial forces load the rod as a column having intermediate supports at each grid support (spacer). The cladding was given elastic-plastic properties and a tensile failure strain of 1.6%, while the fuel was given only elastic properties and no failure strain. Because the fuel has an elastic modulus more than twice that of the cladding and a solid cross-section, almost all of the lateral load resisting capacity ("buckling" strength) of the fuel rod is provided by the fuel, not the cladding. The fuel is basically a coarse granular material with no tensile strength and therefore cannot be relied upon to resist any tensile stress. This natural state of the fuel is not reflected in the applicant's analysis, which assumes that the fuel is a continuous solid with unlimited strength.

This information is needed to satisfy the requirements of 10 CFR 72.236(b), (c), (d), (h) and (I).

3-3 Explain why there is no coating(s) used on the inner cask cavity. If no coating is planned to be used, discuss the amount of hydrogen absorbed during cask immersion and discuss the effects of absorbed hydrogen on potential cracking of the steel.

Alternatively, state that a coating will be used on the inner cask cavity, and demonstrate that the coating to be used is both durable and non-reactive with the cask internal components and fuel elements and remains adherent (20-year license period) when

exposed to the various cask environments. The manufacturer's data/specification sheets referenced in the SAR and test data for these coatings should be submitted to support your argument.

The most prevalent, potentially degrading environments for a cask fabricated from a low-alloy steel include the following: 1) immersion in borated spent fuel pool water during loading and unloading operations, and 2) high temperature and high radiation (including neutrons) environments encountered during vacuum drying evolutions and long-term storage. The statement made on page 1.2-2 of the SAR concerning the inert gas environment is only applicable to storage and not short-term operations.

This information is needed to satisfy the requirements of 10 CFR 72.122(c), and 10 CFR 72.236(h).

3-4 Revise Chapter 3 to provide the mechanical properties applicable to Zircaloy at 60 GWd/MTU burnup along with references for the data submitted.

Based on the reference provided in the SAR, the yield and ultimate strength on page 6A-3 appear to be for low-burnup fuel. Note that the properties will be different for high-burnup fuel. Data is needed to conduct proper structural analysis of the assemblies.

This information is needed to satisfy the requirements of 10 CFR 72.236(a).

3-5 Please re-word or clarify the statement made in Section 3.4.1.4, second paragraph, regarding conditions for hydrogen generation inside the cask.

According to the last sentence of this paragraph, it is implied that the cask is not completely filled with water during loading, which contradicts the cask operating procedures.

This information is needed to satisfy the requirements of 10 CFR 72.11.

Appendix 3E: Fracture Toughness Requirements for Confinement Boundary Material

3E-1 Describe the preheat and post-weld heat treatment (PWHT), if any, and their impact on the fracture toughness properties for the welds. In addition, specify the industry code that will be used to ensure that impact and toughness properties are not compromised during fabrication. No preheat weld treatment was discussed for the shield material in the fracture toughness evaluation.

Should a cask be designed from a low-alloy carbon steel, the air hardening properties of such materials may be a significant adverse factor for the impact properties and fracture toughness. Consequently, for low-alloy steel cask designs, the importance of preheat and post-weld heat treatment (PWHT) is paramount in preventing weld cracking. See Interim Staff Guidance-15, "Materials Evaluation."

This information is needed to satisfy the requirements of 10 CFR 72.236(b).

3E-2 Indicate what acceptance standard/code will be used to repair defects (i.e., in the seam and bottom plate welds) prior to cask use.

This information is needed to satisfy the requirements of 10 CFR 72.122(a), (b), and (c).

Chapter 5 - Shielding Evaluation

5-1 Either revise the application to ensure that the damaged fuel is both essentially structurally intact and that any openings in the cladding would not allow for pellets to fit through or revise the shielding analysis to show dose rates assuming that during the loading and transport to the pad the fuel pellets rearrange such that some pellets fall into the bottom of the storage cask.

This information is needed to satisfy the requirements of 10 CFR 72.104(a).

5-2 Revise the application to show how the cobalt values in Table 5.2-8 were determined using the assembly masses and cobalt weight fractions from Tables 5.2-2 and 5.2-3. The NRC staff's calculations of cobalt values shows significantly more cobalt in each region during the irradiation than does Table 5.2-8. Additionally, it appears from Table 5.3-2 that the amount of cobalt used in the source term calculation was for unchanneled fuel, but channeled fuel will add more source term in the fuel region.

This information is needed to satisfy the requirements of 10 CFR 72.104(a).

5-3 Revise the application to clarify what locations correspond to the dose rates shown in Table 5.4-3. It appears that the dose rates shown in Figure 5.4-1 and Tables 5.4-1 and 5.4-2 should correspond to those shown on Table 5.4-3 for dose rates above the neutron shield. If the dose rates are not at the same location or distance, clarify the application to show the differences between the different dose rate calculations.

This information is needed to satisfy the requirements of 10 CFR 72.104(a).

5-4 Please revise Table 5.3-3 to provide correct values for columns 2 through 4. The values shown in columns 2 through 4 show the same information as column 1.

This information is needed to satisfy the requirements of 10 CFR 72.11 for complete and accurate information.

5-5 Provide references for BWR fuel assembly hardware and spent fuel rod data in Tables 5.2-1a and Table 5.2-2 of the SAR. Specifically, provide a copy of Reference No. 2 of this chapter, "GE Proprietary Fuel Assembly Hardware Data"

The data in these tables are used in structural, thermal, and criticality calculations. The references are needed so that the data can be verified.

This information is needed to satisfy the requirements of 10 CFR 72.11.

Chapter 6 - Criticality Evaluation

6-1 Provide additional clarification and/or justification for Assumption (12) in Section 6.4.1.2. Specifically, provide an explanation as to why the accidents described in this assumption are not considered credible.

This information is needed to satisfy the requirements of 10 CFR 72.236(c).

6-2 Provide reference document, E-21003, Rev. 0, "Design Criteria for the TN-68 Spent Fuel Storage/Transportation Cask for High Burnup and Damaged Fuel."

The document is needed to confirm properties of the fuel, used in the accident analysis, as summarized on page 6B.2-1 and Tables 6B-1 and 6B-2.

This information is needed to satisfy the requirements of 10 CFR 72.11

Chapter 7.0 - Confinement Evaluation

7-1 State in the SAR the basis for the source term identified in Chapter 7, Confinement.

Table 7.3.2, "TN-68 Releasable Source Term for Off-Normal Conditions (Design Basis 8X8 Fuel)" and Table 7.3.3 did not identify the basis for the source term by identifying maximum burnup, maximum enrichment, amount of uranium, cooling time and average power upon which the source term was based.

This information is needed to satisfy the requirements of 10 CFR 72.146(b).

7-2 Change the heading for Table 7.3.3 to be applicable to Hypothetical Accident Conditions.

As submitted, Table 7.3.3 is entitled the same as Table 7.3.2 "TN-68 Releasable Source Term for Off-Normal Conditions (Design Basis 8X8 Fuel)" when it is clearly intended for hypothetical accident conditions.

This information is needed to satisfy the requirements of 10 CFR 72.146(b).

7-3 Adjust the source term shown in Tables 7.3.2 and 7.3.3 to be more representative of the design basis fuel or justify the source term utilized. Include an explanation in the SAR of how the damaged fuel contributed to the source term.

These tables seem to under predict the source term. For example, the staff got higher activities ranging from 20% to 50% for various radionuclides (e.g., Kr-85, Sr-90, Y-90, Cs-134, Cs-137). Source terms should bound the design basis fuel. Staff used the following information to generate the source term: 60 GWd/MTU, 4.7 wt. % U-235 enrichment, 7-year cooling time, 188 kg U/assembly, 60.6 MW/MTU.

This information is needed to satisfy the requirements of 10 CFR 72.146(b).

7-4 Change Note 3 in Tables 7.3.2 and 7.3.3 to be reflective of the design basis fuel and reference Interim Staff Guidance-5, Rev. 1.

This note inappropriately mentions a 7 X 7 BWR array in lieu of an 8 X 8 BWR array, and a 10 year cooling time in lieu of 7 years. Interim Staff Guidance-5, Rev. 1, provides the currently acceptable methods to the staff for performing confinement evaluations. Coincidently, the activity of the Co-60 seems correct for 7 year cooled fuel.

This information is needed to satisfy the requirements of 10 CFR 72.146(b).

7-5 Correct reference in Section 7.3.1 to the table that provides the activities of the various forms of radionuclides.

Section 7.3.1, first sentence makes reference to Table 5.2-5 for a listing of activities of radionuclides contributing more than 0.1%. However, Table 5.2-5 is associated with peaking factor and water density input for determination of axial source distribution.

This information is needed to satisfy the requirements of 10 CFR 72.146(b).

Chapter 8 - Operating Procedures

8-1 Indicate a step in the operating procedure (Table 8.1-1) indicating when and how damaged fuel will be identified.

This step is needed to assure that damaged fuel is placed in the cask only in allowable positions.

This information is needed to satisfy the requirements of 10 CFR 72.11.

Chapter 9 - Acceptance Tests and Maintenance Program

9-1 Indicate, in tabular form, the boron credit, minimum areal density, and the boron carbide volume fraction for each neutron absorber (i.e., Boral®, MMC, and borated aluminum) to be used under this amendment. In addition, include data on previous qualification of the materials to be used under this amendment. Provide the volume fraction qualified and the procedures used for the qualified absorber, as well as details on acceptance plans for each absorber material to be used in the application.

This information is needed to satisfy the requirements of 10 CFR 72.11, 72.24 and 72.236 (c).

9-2 Remove the justification for not conducting thermal and corrosion testing for qualifying a neutron absorber.

The staff has reviewed the literature and some proprietary data in detail, and does agree with the applicant that accelerated radiation testing need not be done on newer absorbers that are made of the same matrix and absorber (B_4C) as previously qualified materials. However, staff does not agree with the applicant that thermal and corrosion

testing should not be conducted. A review of the literature shows that the few tests do not consider synergistic effects of pool chemistry, temperature, galvanic coupling, etc. There is simply not enough published data to unequivocally state that thermal and corrosion testing should not be done under short-term loading operations. The applicant's two technical papers that are referenced to support the argument confirm that there is a paucity of information on thermal and corrosion testing of absorbers. Therefore, whenever the percentage of B_4C exceeds that of previously qualified materials of similar processing and composition, it is required to conduct the tests to establish its durability.

This information is needed to satisfy the requirements of 10 CFR 72.11, 72.24 and 72.236 (c).

9-3 Revise the application to reduce the proposed 90% credit for B10 areal density for Boral®.

Please note that the current SAR reference document for justifying approval of 90% credit, listed as Reference No. 6, is currently undergoing NRC staff review under a separate review schedule. As such, requesting such credit under this amendment request will make its approval contingent on a separate review schedule, which is likely to delay the currently approved schedule. Therefore, the staff recommends that such request be delayed until the staff can resolve the pertinent attenuation and degradation issues associated with credit levels higher than 75% for Boral®.

This information is needed to satisfy the requirements of 10 CFR 72.11, 72.24 and 72.236 (c).

Chapter 10 - Radiation Protection

10-1 Either revise the application to justify determining the estimated dose to workers shown in Table 10.3-1 using the 7 x 7 fuel assemblies instead of the design basis fuel assembly characteristics shown in Section 5.2 and Table 5.2-4, for the 8 x 8 fuel assembly, or revise the application to ensure that the estimated dose to the public is determined using the dose rates calculated in Chapter 5 for the design basis fuel assembly. Note that the doses rates determined in the Shielding Evaluation for the 8 x 8 fuel assembly are higher than those currently used to estimate the dose to workers. Additionally, it appears that surface average dose rates were used to determine the occupational dose instead of location specific dose rates. Revise the application to ensure that the appropriate dose rates are used for each location and activity listed in Table 10.3-1.

This information is needed to satisfy the requirements of 10 CFR 72.104(b) and 72.126.