ENCLOSURE 2

M170085

Draft Responses to the NRC RAIs for the RAJ-II Transportation Package

Non-Proprietary Information – Class I (Public)

IMPORTANT NOTICE

This is a non-proprietary version of M170085 Enclosure 1, which has the proprietary information removed. Portions of the document that have been removed are indicated by white space with an open and closed bracket as shown here [[ ]].

1.0 General Information

Justify the safety classification/category for the fusible plugs and alumina silicate insulation in the Model No. RAJ-II package.

a) Section 1.2.1.2 of the safety analysis report (SAR) describes the use of fire consumable fusible plugs on the outer and inner containers to prevent pressure buildup. SAR Drawings, including 105E3738 (Revision 10), 105E3743 (Revision 7), 105E3745 (Revision 10), 105E3747 (Revision 6), and 105E3748 (Revision 4), identify the fusible plug components with "importance to safety" classifications of "B" or "C" (plug/gasket). Guidance in NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," suggests that the plug components should have an importance to safety classification/category of "A".

b) SAR Section 3.2.1 mentions that the alumina silicate insulation is used "… to repel the rate of heat transfer to the fuel during the fire event." SAR Drawings, including 105E3745 (Revision 10), 105E3747 (Revision 6), and 105E3748 (Revision 4), identify the alumina silicate insulation with a NURE/CR-6407 importance to safety classification of "B". Guidance in NUREG/CR-6407 suggests that the thermal components, including insulation, should have an importance to safety classification/category of "A".

This information is needed to determine compliance with 10 CFR 71.31(a)(1), 71.33.

<u>GNF Response</u>

1.a) While the importance to safety classification was added to the drawings in this revision of the SAR, the classification itself has not changed. The original classification document, GNF document 0000-0034-3649, dated November 19, 2004, sets the justification behind this classification by stating that failure of the fusible plugs "has no impact on containment or criticality." To clarify further on this rationale, the definition of a Category A item in NUREG/CR-6407 is any item "whose failure could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry comprising criticality control" (Section 3, Table 2). As noted in the RAI, the purpose of the fusible plugs is to limit pressure increase during accident conditions, and, as such, they could be considered pressure relief devices, which are listed as Category A items in Section 5.3 of the NUREG. However, the expected failure of these plugs is not on the primary containment, which is the fuel rod. Loss of the containment upon their failure is not a concern. Even in the unlikely event that these plugs would fail to melt, the packaging is not air tight (see SAR Section 2.12.4, for example) and, as such, pressure would be self-limiting. Therefore, there is no concern that a malfunction would significantly reduce the packaging effectiveness, creating a

situation adversely affecting public health and safety. These components are correctly classified as Category C.

1.b) While the importance to safety classification was added to the drawings in this revision of the SAR, the classification itself has not changed. The original classification document, GNF document 0000-0034-3649, dated November 19, 2004 sets the justification behind the classification of the alumina silicate insulation by stating that "multiple failures could reduce FA spacing." Per NUREG/CR-6407, a Category A item is any item "whose failure could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry comprising criticality control" (Section 3, Table 2). Since the original justification states that multiple failures are required for any impact, the definition is correctly stated as a Category B.

3.0 Thermal

3-1 Provide a quantitative analysis to demonstrate that the package, with the cumulative effect of the 10 CFR 71.73 hypothetical accident conditions, would not exceed the regulatory release.

SAR Section 2.7.4 stated that the maximum hypothetical accident condition testing temperature for an earlier-designed fuel assembly was 921 K (1198°F) and that the fuel rod pressure due to accident conditions does not exceed 508 psig (522.7 psia). The SAR also stated that the fuel rods have a rupture pressure in excess of 520 psi (pressure value was not provided). It was not evident that the condition (e.g., strain) of the modified fuel assemblies' deformed fuel rods, after the hypothetical accident condition 30 ft drop (end drop, side drop, etc.) and puncture tests, was quantitatively considered when analyzing at the high temperatures and pressures (e.g., 1198°F and 508 psig) of the thermal hypothetical accident condition and during subsequent calculations that are used for input in the certificate of compliance fuel parameter tables.

This information is needed to determine compliance with 10 CFR 71.51(a)(2) and 71.73.

<u>GNF Response</u>

Section 2.7.4.1 of NEDE-33869P (Reference 3-1.1) states that the maximum predicted HAC temperature for any fuel type is 648°C (1198°F), which is below the allowable HAC temperature limits presented for the various fuel types. Section 2.7.4.1 of Reference 3-1.1 also states that the maximum predicted HAC pressures do not exceed the allowable limits listed in Section 3.0. Section 3.1.4 defines the maximum predicted pressure for BWR fuel as 508 psia, which is below its allowable pressure presented in Section 3.4.3.2.1. In summary, Reference 3-1.1 demonstrates that all fuel types remain below both temperature and pressure limits required to maintain containment.

Reference 3-1.1 discussions regarding fuel rod performance during a HAC event do not detail the impact of strain experienced by the fuel rods of the dropped certification test unit assemblies described in Table 2-11 of that reference. Therefore, additional analyses are presented below.

Of all the RAJ-II drop test orientations, the 90° end drop test of CTU-2 was the only test article where the fuel rods experienced visible deformation, as described in Table 2-11. For CTU-2, the fuel rods had no significant damage but did experience minimal deformation localized between the lowest two spacers adjacent to the lower tie plate, as shown in Figure 2-41 and 2-43 of Reference 3-1.1, and reproduced in Figure 3-1.1 and Figure 3-1.2 within this response.

Based on the radius of curvature of the most deformed fuel rod, shown in Figure 3-1.1 and Figure 3-1.2 as the rod closest to the exterior of the inner container, the maximum strain at the rod outer diameter, c of 0.202 inch, is estimated to be less than 5% strain. The estimated strain is calculated based on a deflection, y, conservatively assumed as the width of two fuel rods [[ ]] over an assumed [[ ]] span length, l. These estimations are taken from the deformation depicted in Figure 3-1.1 and Figure 3-1.2. In reality, the fuel rod

deflection is much less than assumed, and the span length is greater than assumed; therefore, this is a conservative approach in determining the radius of curvature. A deflection of this amount results in a radius of curvature, ρ, of [[ ]].

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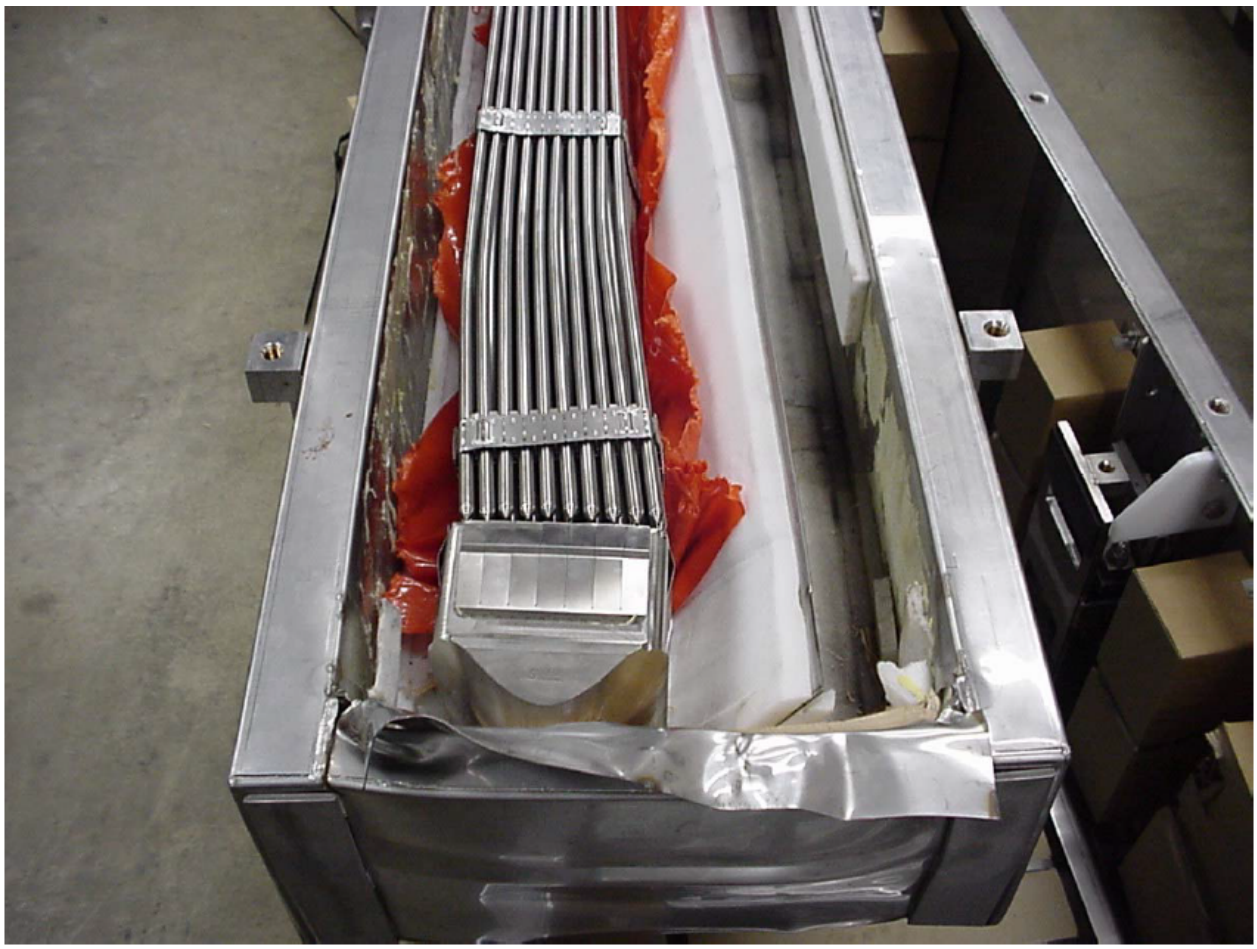

Figure 3-1.1: CTU-2 Post 9m Impact

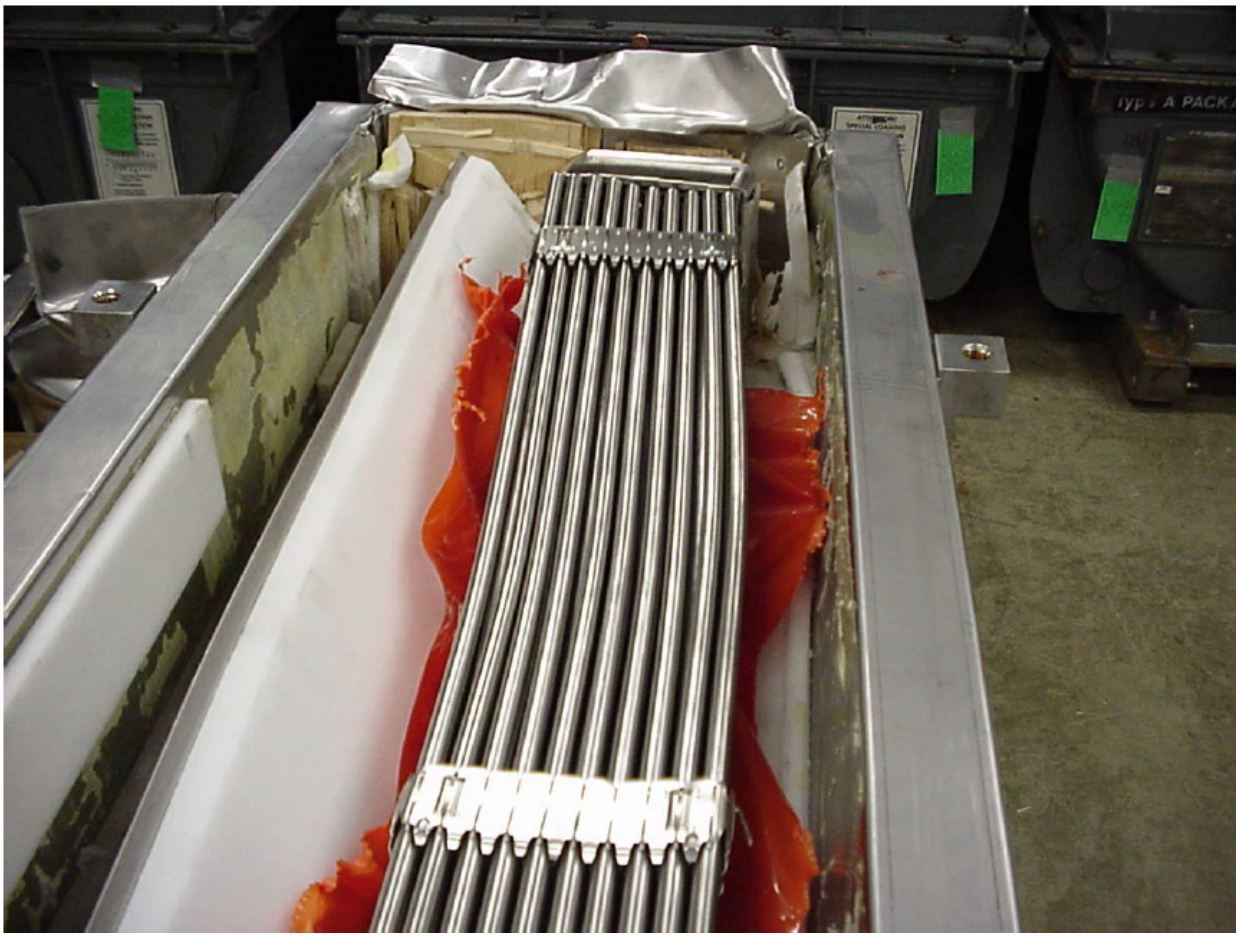

Figure 3-1.2: CTU-2 Post 9m Impact

Based on Figure 3 of Reference 3-1.2, there is negligible impact on the mechanical performance, strength, and ductility of zirconium alloys for less than 5% strain. Since the plastic strain associated with the 9m impact drop test of the RAJ-II produces less than 5% strain in the fuel cladding, the cold working associated with the impact will not affect the fuel rod thermal performance during the HAC fire event.

During fabrication, zirconium cladding is cold-worked and annealed, as seamless tubing is manufactured by drawing, extruding, or pilgering. Tubes manufactured with these processes experience a high percentage of cold work and are subsequently annealed at temperatures typically less than 700°C. Since the manufacturing anneal temperature is below the estimated fuel rod rupture temperatures presented in Section 3 of Reference 3-1.1, the effects of cold working are further reduced.

In summary, based on the small degree of cold work experienced by the fuel during the 9m drop, and because the subsequent fire event temperature of 800°C is above the anneal temperature for the zirconium alloy tubing, the effects of less than 5% strain are negligible on the thermal performance presented in Reference 3-1.1. This is confirmed by real events discussed in Reference 3-1.3. This reference discusses a severe accident, where a passenger vehicle traveling the wrong way down the road impacted a tractor trailer carrying 24 unirradiated nuclear fuel

assemblies with zirconium alloy cladding. In this event, the car collided with the truck, and then the truck collided with a guardrail. A fire started due to the collision, and some fuel at the front of the trailer not only experienced extreme temperature conditions above HAC, but also fell from the trailer bed. Though this fuel incurred plastic deformation in excess of the RAJ-II 9m drop CTU-2 assembly, the fuel thermal performance remained consistent with that outlined in Reference 3-1.1.

References

3-1.1 Letter, Scott P. Murray (GNF) to Director, Division of Spent Fuel Management (NRC), "GNF-A Request for Revision of Certificate of Compliance (CoC) USA/9309/B(U)F-96 for Model No. RAJ-II Package," SPM 16-035, September 30, 2016.

3-1.2 "Influence of Alloy Composition on Work Hardening Behavior of Zirconium-Based Alloys," Hyun-Gil Kim, Il-Hyun Kim, Jeong-Young Park, and Yang-Hyun Book, LWR Fuel Technology Division, December 26, 2012.

3-1.3 NUREG/CR-5892, "A Highway Accident Involving Unirradiated Nuclear Fuel in Springfield, Massachusetts, on December 16, 1991."

3-2 Provide the analysis that demonstrates that the benchmarked creep test data/model can be used to evaluate the creep thermal performance criteria for BWR ($r/t|_{BWR} * P_f * 921/293 - P_a) \leq 31.1$ MPa (4,514 psi)) and non-BWR ($r/t|_{non\text{-}BWR} * P_f * 921/293 - P_a) \leq 56.3$ MPa (8,166 psi)) fuel.

The response to RSI 3-1 indicated that some of the fuel designs used in the RAJ-II package were outside the range of the creep test parameters. However, there was no analysis provided that demonstrated the validity of the creep tests results/model for determining the BWR and non-BWR allowable stress and thermal stress criteria presented in SAR Sections 3.4.4.1 and 3.4.4.2.

This information is needed to determine compliance with 10 CFR 71.51 and 71.73.

GNF Response

The thermal performance of the BWR and non-BWR fuel rods under high stress creep is evaluated using cladding creep relations and material properties from the PRIME03 (Reference 3-2.1) nuclear fuel performance code. PRIME03 is an NRC approved methodology for fuel thermal mechanical analysis (TAC # MD4114). The relations are extracted from PRIME03 and computed to assess clad creep rupture as a function of temperature with rod internal pressure as a dependent variable with the required 30-minute hold time to reflect the length of the Hypothetical Accident Conditions (HAC) external fire event. As a validation of this analysis, GNF2 clad dimensions are utilized and compared to the autoclave testing performed by GNF on GNF2 rods. The comparison to the test data demonstrates the validity of the model employed. Various postulated rod geometries and fill pressures are then chosen to determine the maximum temperature below which failure is not observed which corresponds to a given hoop stress at the event initiation. The results of this study demonstrate that the failure temperature is a strong function of the hoop stress at event initiation and is independent of other parameters, largely due to the fact that the hoop stress encapsulates important parameters such as geometry and fill pressure. This result validates employing the model for different rods with different geometries at different fill pressures.

Evaluation Method

Creep is a plastic deformation mechanism that involves temperature-dependent atomistic movement. There are three general creep mechanisms. [[

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Creep models used are consistent with those defined in Section 5.6 of PRIME03 (Reference 3-2.1). [[

]] In this analysis, the stress that results in creep is the cladding hoop stress that is set by the rod internal pressure. [[

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Zircalloy material properties are used in all analyses. [[

]] Initial Pressure is taken at 26°C and then adjusted using the ideal gas law (PV=nRT) to compute the initial pressure at temperature. The creep strain limit is taken as [[

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Evaluation Results

Creep calculations are performed to determine the failure temperature of GNF2 fuel rods to compare to the test data. GNF tested BWR fuel rods in an autoclave. The GNF2 rods were heated to temperatures in increments of 50°C where the rods were held at the temperature for approximately 30 minutes prior to commencing the 30-minute test to ensure they were at temperature. At 800°C, failure was not observed. [[

]] The same rod was modeled using the evaluation method described above. The results of that analysis are shown in Table 3-2.1.

Table 3-2.1 GNF2 Rod Internal Overpressure and Hoop Stress for Failure during the Event

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The predicted failure temperature for the GNF2 rod was [[ ]]. This is very close to the measured failure temperature which was determined to be between 800 and [[ ]]. The now validated creep model is applied to the PWR rod using the same model. The results of this analysis are shown in Table 3-2.2.

Table 3-2.2 PWR Rod Internal Overpressure and Hoop Stress for Failure during the Event

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It is observed that the PWR rod has a [[
]]. There is a strong correlation between stress at event initiation and failure temperature, regardless of specific rod geometry or initial pressure. This is due to the fact that the hoop stress at event initiation captures the initial pressure as well as the rod geometry through radius to thickness ratios; therefore, the hoop stress is an applicable criterion to use across different fuel geometries. As the hoop stress at event initiation is highly correlated with failure temperature, the criterion is broadly applicable across different fuel geometries and initial pressures. In order to demonstrate the robustness of the model, different postulated fuel rod geometries, initial fill pressures and failure temperatures were analyzed to determine the maximum hoop stress at the event initiation. A strong correlation [[ ]] was observed between the hoop stress at event initiation and failure temperature, regardless of rod type, fill pressure or failure temperature. The rods modeled are shown in Table 3-2.3 while the correlation is shown in Figure 3-2.1.

Table 3-2.3 Postulated Rod Geometries, Fill Pressures and Temperatures to Determine Hoop Stress Predictions

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Figure 3-2.1 Hoop Stress at Event Initiation as a Function of Failure Temperature

As shown in Figure 5.18 of NUREG-5892 (Reference 3-2.4), zirconium alloys undergo a phase transition from hexagonally close packed (α phase) to body centered cubic (β phase) at approximately 815°C, where the phase transition completes at approximately 980°C. This β phase is weaker than the α phase. [[

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Table 3-2.3 and Figure 3-2.1 demonstrate that the validated creep model from Reference 3-2.1 and described above predicts failure temperatures accurately for a variety of fuel rod geometries and initial pressures based on the accurate computation of the hoop stress at event initiation.

References

3-2.1 "The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance Part 1 – Technical Bases," NEDC-33256P-A Revision 1.

3-2.2 “TRACG Model Description”, NEDE-32176P Revision 4.

3-2.3 NUREG-0630, “Cladding Swelling and Rupture Models for LOCA Analysis”.

3-2.4 NUREG-5892, “A Highway Accident Involving Unirradiated Nuclear Fuel in Springfield, Massachusetts, on December 16, 1991”.

4.0 Containment

4-1 Specify the minimum cladding thickness in Table 4, "Fuel Rod Parameters," of the draft Revision 11 of the CoC proposed by the applicant.

In the draft proposed Revision 11 of the CoC, the applicant requested removing the cladding thickness from Table 4, "Fuel Rod Parameters." In contrast, CoC Sections 5(b)(1)(iv) and (v) both reference the minimum clad thickness in Table 4. When comparing the Fuel Rod OD and Cladding ID parameters in Table 4, the fuel could effectively have a cladding thickness of zero. Since the fuel cladding is part of the containment boundary, removal of the cladding thickness parameter and the possibility of a zero clad thickness is not appropriate. The fuel cladding thickness in Table 4, "Fuel Rod Parameter," and Table 3, "Fuel Assembly Parameter," should reflect the values that have been demonstrated to meet structural, thermal, and containment performance during NCT and HAC.

This information is necessary to determine compliance with 10 CFR 71.51(a) and (b).

GNF Response

The minimum cladding thickness will be returned to the proposed CoC exactly as shown in Revision 10 of the CoC.

7.0 Operating Procedures

7-1 Revise the title for Table 7-1, "Recommended Packaging Component Torques," to, "Required Packaging Component Torques." Revise Table 7-1 to provide a plus or minus ft-lb torque value next to all the torque values listed in the torque column. Revise the Table 7-1 note and the text throughout Chapter 7 from "recommended torque" to "required torque."

This information necessary to determine compliance with 10 CFR 71.87(c).

GNF Response

The requirements of 10CFR 71.87(c) state that "each closure device of the packaging, including any required gasket, be properly installed and secured and free of defects." Previous revisions of the SAR used the wording "wrench tight or as defined by procedures" to describe the bolt tension needed for the RAJ-II outer container lid bolts, the inner container lid bolts, and the hold-down bar bolts, of which none are containment boundaries of the package. This revision sought to clarify previous wording by adding the torque from the applicable procedures as a recommended torque guidance due to the vagueness of "wrench tight." However, it intentionally specified these values as recommended guidance and not required torque since a specific bolt torque is not required for any safety function, for example SAR Section 2.12.4.