AREVA

Ref: AFS–16–0036

January 26, 2016

ATTN: Document Control Desk
Director, Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555–0001

SUBJECT: 1) BRR Package Amendment Request, Docket No. 71–9341, CAC No. L25031

2) Letter from Norma Garcia-Santos to Philip W. Noss of November 10, 2015, transmitting NRC request for additional information (RAI)

AREVA Federal Services LLC (AFS) hereby submits Revision 9 of the Safety Analysis Report for the BRR Package, Docket No. 71–9341. This SAR revision includes several changes made in response to the RAI received by AFS on November 10, 2015.

Responses to each request for information are provided in Attachment A. The principal change consists of revising the material of the loose plate box and Square fuel pedestals from aluminum to stainless steel, and taking credit for the box in the shielding and criticality evaluations. Accordingly, a revised basket drawing, 1910-01-03-SAR, has been provided. In addition, changes to SAR Chapters 1, 2, 3, 5, 6, and 7 are included. These chapters will show Revision 9 in the header of each page, and each change is marked by a revision bar in the right page margin. SAR Chapter 4, Section 2.12.2 through Section 2.12.7, and Chapter 8 have not been revised and will show Revision 8 in the header of each page.

In addition, the following minor changes are included in Revision 9 of the SAR:

1. Drawing 1910-01-03-SAR, General Note 5, has been revised to remove aluminum welds from, and to include intermittent welds in, the weld types which are excepted from the liquid penetrant inspection requirement. Aluminum is no longer used as a material. Intermittent welds should not be inspected by liquid penetrant means since the start and stop of the welds may show a false indication reading.
2. General Note 19 has been added to drawing 1910-01-03-SAR to permit markings to be placed on any of the baskets to aid in verifying proper placement of fuel elements during loading. These markings have no effect on any safety evaluation.
3. Drawing 1910-01-03-SAR has been revised to show a changed feature for lifting the loose plate box. The prior design included two holes in two opposite side plates of the box. It was concluded that these holes might be difficult to use if the box is filled with material. The new lifting feature consists of trunnions, located essentially where the prior holes were located. The trunnions will be used with a small lifting yoke in a licensed facility. This change has no effect on any safety evaluation.
4. The PULSTAR gamma and neutron source terms and decay heat have been recomputed using a larger reactor power to reflect an upcoming power uprate at the PULSTAR reactor.

Included with this letter is one paper copy of the SAR and one CD-R containing the PDF file "BRRC SAR, Complete, Rev. 9.pdf" (30,734 kb, 651 pages). The CD is contained within an envelope labeled, "BRR Package, Docket 71–9341 SAR Revision 9, Electronic Copy of Document". To update a paper copy of the SAR, replace the cover sheet, Table of Contents, and Chapters/Sections 1, 2, 2.12.1, 2.12.8, 3, 5, 6, and 7 in their entirety. The other parts should be left as Revision 8.

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Should you have any questions regarding this submittal, please contact me at (253) 552–1321 or via E–mail (phil.noss@areva.com).

Very Truly Yours,

AREVA Federal Services LLC

Phil Noss

Phil Noss
Licensing Manager

cc: Ms. Norma Garcia-Santos, NRC (including one paper copy and one CD)
Mr. Douglas Morrell, DOE NEUP (including one paper copy and one CD)

ATTACHMENT A

AREVA Federal Services, LLC Responses to NRC Request for Additional Information Dated November 10, 2015

Docket No. 71-9341
Certificate of Compliance No. 71-9341
Model No. BEA Research Reactor (BRR) Package

The following material contains the response of AREVA Federal Services LLC (AFS) to the Request for Additional Information of the NRC dated 11/10/2015. Each response is preceded by the NRC question.

GENERAL

G-1. Demonstrate that the loose plate fuel basket will not deform and the fuel will not move in the box under hypothetical accident conditions or provide safety analyses with appropriate justification for the assumptions about the degree of deformation. The demonstration should include the following information as this relates to the containment evaluation, shielding evaluation, and criticality safety evaluation:

a. releases of radioactive material with bounding fuel plate damage resulting from loose plate fuel basket deformation;

b. impact on dose rate at 1 meter from the surface of the package under hypothetical accident conditions with justified change of source geometry; and

c. the impact on the k_{eff} of the package under hypothetical accident conditions with justified fuel geometry.

The applicant proposes nearly square or slightly curved loose fuel plates as authorized contents for the Model No. BRR. The request includes approval of a new basket design for shipping these fuel plates in the Model No. BRR packaging system. However, the applicant did not provide structural stability analyses demonstrating that the loose fuel basket will not deform or the fuel will not be damaged to cause fuel deformation or cladding breach.

This information is needed to determine compliance with 10 CFR 71 .31 (b), 71.73(c), and 71 .55(d).

Response: The analysis of the loose plate box has been revised to show that the box does not deform under NCT or HAC, and the possibility for movement of the fuel plates will be restricted. Drawing 1910-01-03-SAR has been revised to change the material used for the loose plate box and for the associated spacer pedestals from aluminum to ASTM Type 304 stainless steel. Section 2.12.8 has been revised to show that the loose plate box and spacer pedestals will retain their structural integrity under the worst case HAC loading. Thus, the loose plates will remain within the confines of the box during transport. Section 7.1.2.1 and Section 7.1.2.2 have been revised to require the insertion of aluminum dunnage into the loose plate box beside the fuel plates to limit the potential for motion of the loose plates during transport. The criticality evaluation in Chapter 6 has been revised to include the presence of the loose plate box under HAC. The shielding evaluation in Chapter 5 has been revised to change the box material from aluminum to stainless steel.

CRITICALITY SAFETY

Cr-6-1. Provide detailed drawing(s) for the loose plate basket demonstrating that the loose plates will remain evenly distributed during normal conditions of transport (as modeled).

The applicant states the following in the application:

> "A loose plate box is used to transport up to 31 loose plates per basket. Loose plates are limited to U-Mass (aluminide), U-Florida, and Purdue fuel plates."

However, the Licensing Drawing No. 1910-01-01-SAR does not show the location, spacing, and distribution of loose plates in the basket. In addition, this licensing drawing shows that the internal cavity of the loose plate basket is 7.62 centimeters (cm) long and 6.35 cm wide.

Based on the information provided in Table 6.2-11 of the application, the "specifications" for the U-Florida fuel plates are as follows:

i. 6.2738 cm wide,
ii. 0.0762 cm thick, and
iii. the channel spacing is 0.29718 cm.

The total thickness of the 31 plates would be as follows:

i. 2.3622 cm without channels, and
ii. 11.57478 cm with channels.

As such, 31 loose plates <u>with channels</u> would not fit into a loose plate basket.

For 31 or fewer plates <u>without channels</u>:

i. the fuel plates would take less than one third of the space of the loose plate basket cavity, and
ii. there will be a 5.62 cm space to allow the plates to move in a random manner.

This information is needed to determine compliance with 10 CFR 71.55.

Response: The criticality analysis of the loose plate box has been revised based on the changes discussed in the response to RAI G-1. Credit is now taken for the geometry control provided by the box. Aluminum plate dunnage is now required in the box to minimize the void space, although this dunnage is not credited in the criticality analysis. Uniform pitches, non-uniform pitches, and 31 or fewer plates are considered in the analysis.

Note that U-Florida plates have 0.112-in thick spacers welded to the plates and therefore cannot pack tightly. The loose plate box cannot fit more than approximately 17 U-Florida plates. It is demonstrated in the criticality analysis that reactivity is maximized with 31 plates. In contrast, U-Mass (aluminide) loose plates and Purdue loose plates have no spacers attached, and the plates can rest in close contact.

Cr-6-2. Demonstrate that a Model No. BRR package loaded with 31 loose fuel plates (maximum number of plates per basket) meets the regulatory requirements of 10 CFR 71.55(d)(2).

On page 1.2-9 of the application, the applicant states the following:

> "A loose plate box is used to transport up to 31 loose plates per box. Loose plates are limited to U-Mass (aluminide), U-Florida, and Purdue fuel plates."

Therefore, the package may be loaded with fewer than the maximum allowable content.

Also, from the Licensing Drawing No. 1910-01-01-SAR and the fuel plates geometry data, there will be a large space in the loose plate fuel basket, even if the basket is loaded with the maximum payload of 31 loose fuel plates without channels. The gap between plates will be even larger if the basket is loaded with fewer loose plates. It is not clear how the package meets the requirements of 10 CFR 71.55(d)(2) because the content may move around during loading and transportation.

This information is needed to determine compliance with 1 O CFR 71.55(d)(2).

Response: The criticality analysis of the loose plate box has been revised based on the changes discussed in the response to RAI G-1. Credit is now taken for the geometry control provided by the box. Aluminum plate dunnage is now required in the box to minimize the void space, although this dunnage is not credited in the criticality analysis. Uniform pitches, non-uniform pitches, and 31 or fewer plates are considered in the analysis.

Cr-6-3. Revise the application to ensure that the package meets the criticality safety requirements of 10 CFR 71.55 and 71.59 for less than full loads of loose fuel plates, and for plates unevenly spaced within the basket. Revise the criticality safety index for loose fuel plates in the Model No. BRR package accordingly.

The applicant states that the loose plate box is intended to be used to transport *up to 31* plates per box. Therefore, the package may be loaded with fewer than the maximum number of plates. Additionally, the criticality analysis for loose fuel plates assumes equal spacing between the plates, even though the plates are loose with no structural material to ensure this spacing.

Given that the system is under-moderated when the package is fully loaded, as discussed on page 6.3-3 of the application, the package with fewer loose fuel plates, or with uneven spacing between plates, could be more reactive. The application should be revised to demonstrate that packages loaded with fewer than 31 loose fuel plates per box, and boxes with uneven spacing between loose fuel plates, remain subcritical. This demonstration should include single packages with water inleakage per 10 CFR 71.55(b), single packages under normal conditions of transport per 10 CFR 71.55(d), single packages under hypothetical accident conditions per 10 CFR 71.55(e), and arrays of packages under normal conditions of transport and hypothetical accident conditions per 10 CFR 71.59.

This information is needed to ensure compliance with 10 CFR 71.55 and 10 CFR 71.59.

Response: The criticality analysis of the loose plate box has been revised based on the changes discussed in the response to RAI G-1. Credit is now taken for the geometry control provided by the box. Aluminum plate dunnage is now required in the box to minimize the void space, although this dunnage is not credited in the criticality analysis. Uniform pitches, non-uniform pitches, and 31 or fewer plates are considered in the analysis.

Note that the HAC single package analysis, which includes optimum moderation, bounds the requirements of 71.55(b).

MATERIALS EVALUATION

M-1-1. Describe in detail how the applicant or user determines the extent of damage for PULSTAR, TRIGA, and Square plate fuels during hypothetical accident conditions.

Except for visual inspection of the fuel , the applicant does not describe an operation or process to determine the magnitude of the damaged for each type of fuel during hypothetical accident conditions (see Sections 2.2, 2.7.8, and 3.2.1 of the application). The applicant (or user) monitors:

i. excessive corrosion/erosion,
ii. mechanical wear and damage, and
iii. plate swelling (and/or blistering).

The applicant assumes the following in its analysis for damage fuel under hypothetical accident conditions:

i. square plate fuels keep their structural integrity with conservative removal of end structure (see Section 6.3.1 of the application),

ii. PULSTAR fuel is similar to the commercial pressurized water reactor (PWR) fuel (see Section 6.3.4 of the application), and
iii. PULSTAR fuel expands until constrained by the Zircaloy fuel box (see Section 6.3.1 of the application).

The conservatisms and the basis for their use in the evaluation of fuel damage for the proposed contents are unclear [i.e ., reference damage (e.g. , damages shown on page 3.5-3 of the application, Reference No. 30)]. The damage information referenced was for a different fuel (i.e., aluminum spent fuel).

This information is needed to determine compliance with 10 CFR 71.43(d) and 71.73(c).

Response: Section 2.7.1.8 has been added to the Safety Analysis Report. This section evaluates the behavior of plate-type fuel, TRIGA fuel, and PULSTAR fuel in the bounding HAC impact orientations. The evaluation shows that all fuel types and varieties, with one exception, retain their structural integrity in the bounding HAC impact. The exception is U-Florida plate fuel. This fuel is installed with a spacer plate to limit the free space available within the basket opening (see Section 7.1.2.1 and Section 7.1.2.2), and the criticality evaluation for this fuel is performed using worst-case assumptions regarding plate spacing (see Section 6.4.1.2). The results of these evaluations are:

a) The behavior of all fuels is demonstrated to support the criticality geometry assumptions made in the criticality evaluation
b) Where fuel integrity could not be shown (U-Florida fuel), free space is restricted by insertion of a spacer at the time of fuel loading, and the criticality evaluation assumes worst-case spacing in the remaining space
c) All k_{eff} remains well below the USL.

M-1-2. Provide the basis for the assumption that the release of fission-generated gases for aluminum-based uranium dioxide (UO_2) fuel is applicable to the types of fuel proposed as authorized contents in this revision to the Model No. BRR.

In Section 3.1.4 of the application, the applicant mentions that the release of fission-generated gases from uranium-aluminide and uranium-zirconium hydride based fuels is diffusion-limited to the PULSTAR fuel and insignificant:

i. 6 pounds per square inch gauge [lb_f/in^2 gauge or psig (NORMAL CONDITIONS OF TRANSPORT limit, 10 psig)], and
ii. hypothetical accident conditions 12 psig with 2.9 psig fission gas release (maximum pressure of 25 psig).

The applicant assessed a diffusion-limited slow gas release with aluminum based U02 fuel (see page 3.5-3 of the application, Reference 30). The applicant did not provide the basis for assuming that this assessment is applicable to the other types of fuel under consideration.

This information is needed to determine compliance with 10 CFR 71.43(d).

Response: The release of fission gases from the various types of payload fuels is discussed in Section 3.3.2.

- The release of gas from plate-type fuels is based on the discussion in Reference 30 of Chapter 3, and remains unchanged.
- The release of gas from TRIGA fuel was based on Reference 31 of Chapter 3, which was a page from the General Atomics website. To better support the assertion that release of fission gas from TRIGA fuel is negligible, Reference 31 has been changed from the General Atomics website to a paper entitled, "Higher Power Density TRIGA Research Reactors", by W. L. Whittemore, General Atomics, San Diego, CA.
- The release of gas from PULSTAR fuel was discussed in Section 3.3.2 of Revision 8 of the SAR under the heading "Fission Gas Release from PULSTAR Fuel". This discussion relied on the technique outlined in Reference 39 of Chapter 3. For Revision 9 of the SAR, the section on PULSTAR fuel has been slightly revised to clarify the discussion, but the calculations and results do not change from Revision 8 of the SAR.
- Finally, Section 3.1.4 has been revised to better describe the results obtained in Section 3.3.2.

M-1-3. Describe the vacuum drying process after loading PULSTAR, TRIGA, and Square Plate undamaged and damaged fuels. Also, explain how the applicant ensures that residual water is not present in the package (after the drying process) in order to avoid adverse chemical reactions.

In Section 3.3.3, the applicant mentions that under inadequate drying conditions, the fuel can generate gases by radiolysis. The staff notes that in humid conditions and a long period of time, galvanic corrosion (e.g., aluminum and stainless steel) may also occur.

This information is needed to determine compliance with 10 CFR 71.43(d).

Response: The drying process used with the BRR package is outlined in Section 7.1.2.1, Steps 33 through 39, and in Section 7.1.2.2, Steps 29 through 35. Note that both the wet loading procedure and the dry loading procedure include a series of steps to ensure the dryness of the cask cavity prior to shipment. These procedures utilize a vacuum and dwell time to evacuate any residual moisture from the cask cavity. The criteria for dryness is that the pressure within the cask must not exceed 3 Torr after 30 minutes of isolation from the vacuum pump. Subsequently, the cask cavity is backfilled with helium. Gas generation from radiolysis, or corrosion of the fuel, will not occur under these conditions.

SHIELDING EVALUATION

Sh-5-1. Explain the assumption for the mass of stainless steel in the stainless steel rods with a length greater than 30 inches.

On page 5.2-6 of the application, the applicant states the following:

> "From [4], the mass of stainless steel in the stainless steel clad standard rod (Type 103) is 800 g. This rod is 29.15-in long. It is assumed that all TRIGA rods with a length < 30-in will have 800 g stainless steel. The

> longest rods have an overall length of 45.5-in. Due to the increased cladding length, it is assumed that all rods with a length > 30-in have 1000 g stainless steel."

The mass of a 30-inch long stainless steel rod is approximately 800 grams. The longest rod has a length of 45.5 inches. Therefore, assuming that these rods each have a mass of 1,000 grams is not realistic. A 45 inch rod should have a mass greater than 1,000 grams, based on the stainless steel mass per unit length of the 30-inch rod.

This information is needed to determine compliance with 10 CFR 71 .33(b)(1).

Response: The mass of stainless steel in a TRIGA element is split between the cladding and end caps. When comparing a 30-in long TRIGA element with a 45-in long TRIGA element, the mass of stainless steel only increases due to the increase in cladding length, as the end cap mass does not change.

The total stainless steel mass of a Type 103 element is 800 g. Dimensional data for this TRIGA element is provided in Table 5.2-7 of the SAR. This element has an OD of 1.48-in, a cladding thickness of 0.02-in, and a length of 29.15-in. The stainless steel density is 7.94 g/cm^3.

The longest element is 45.5-in in length, or an additional 16.35-in compared to Type 103. The mass of stainless steel in 16.35-in of cladding is 195 g. Therefore, the total stainless steel mass of the longest element is approximately 800 g + 195 g = 995 g, which is rounded up to 1000 g stainless steel.

Note that these masses refer only to the mass of stainless steel in the element, not the total element mass. The total mass of an element is larger than the stainless steel mass. The mass of stainless steel is of interest for shielding due to Co-60 activation.

Sh-5-2. Explain the rationale for the assumptions used to model pitch for all TRIGA fuels including selecting IPR-RI TAIGA fuel as the bounding case and not other types of fuel. Also, provide a description of the type(s) of IPR-RI TAIGA fuel used for modeling pitch for all TRIGA fuels.

On page 5.2-6 of the application, the applicant states the following:

> "The fuel element pitch used in the models is estimated based on data for the IPR-R1 TRIGA reactor. IPR-R1 has an effective triangular pitch of 4.404 cm and a TRIGA element with an outer diameter of 3.76 cm. Assuming the ratio of pitch (P) to rod diameter (D) is approximately constant for any host reactor, P/D = 4.404 cm/3. 76 cm = 1 .171, or P = 1.171 D. This pitch is used on the LATTICECELL card for each rod diameter D. When building the NEWT geometry model, the outer boundary is modeled as a square because a hexagonal outer boundary is not allowed, and the dimension of the square is selected to preserve the area of the unit cell. The area of the unit cell is $\sqrt{3}/2$ P^2."

Describe the characteristics of IPR-RI and why the applicant selected this specific fuel as the bounding case. In several places, the applicant mentions that TRIGA Type 109 fuel or Type 219 fuel was used as the bounding condition for this licensing action.

This information is needed to determine compliance with 10 CFR 71.47(b).

Response: TRIGA elements from the IPR-R1 TRIGA reactor will not be shipped in the BRR package. IPR-R1 TRIGA reactor data was used to estimate the element pitch simply because this data was publicly available and TRIGA reactors are highly standardized. To facilitate response to this RAI, General Atomics, the TRIGA reactor designer, has provided TRIGA design information to AFS. The TRIGA shielding analysis has been revised using this data. In the revised analysis, a smaller pitch is used in the TRITON models, consistent with the minimum pitch that is available in a TRIGA reactor (the previous models had used an average pitch). Reducing the pitch reduces the amount of water in the lattice, which leads to a larger neutron source. As a result of this change, the neutron dose rate from TRIGA fuel has increased but remains significantly below the dose rate limits.

Sh-5-3. Provide the following information about the bounding conditions used for evaluating the proposed contents:

a. Explain why the TRIGA Type 219 is not modeled explicitly as Type 109;
b. Clarify which TRIGA fuel type (109 or 219) is used as bounding condition; and
c. Provide supporting information to demonstrate that TRIGA Type 109 or 219 is bounding for all enrichments, burnups, and cooling times requested in this licensing action.

On page 5-2-6 of the application, the applicant states the following:

> "This MCNP model is described in detail in Section 5.3, *Shielding Model.* In the MCNP model, element Type 109 is modeled explicitly and the source is distributed evenly throughout the fuel matrix. Element Type 109 is modeled in MCNP because it has the highest U-235 enrichment (70%) of all TRIGA elements and the criticality analysis demonstrated it is the most reactive ."

On page 5-2-7 of the application, the applicant states the following:

> "The dose rates computed at the side of the cask for each TRIGA source are provided in Table 5.2-8. The maximum package surface dose rate of32.2 mrem/hr occurs for Type 219 with a burn up of 119 MWD and a cooling time of 530 days burnup. TRIGA Type 219 has the largest uranium loading of all TRIGA elements considered (825 g U) and the largest burnup (in MWD)."

It is not clear from the cited text which assembly type is considered bounding in the shielding analysis, and why.

This information is needed to determine compliance with 10 CFR 71.47(b).

Response: TRIGA Type 219 has the largest source term and is bounding for shielding, as documented in Chapter 5, *Shielding Evaluation*. TRIGA Type 109 is the most reactive element type and is bounding for criticality, as documented in Chapter 6, *Criticality Evaluation*. In the MCNP shielding models, a hybrid approach is employed, where the bounding Type 219 source (gamma/s and neutrons/s) is modeled but the pellet material is modeled with the Type 109 material composition.

This hybrid approach was employed simply as an additional conservatism. The TRIGA fuel is modeled with a fresh pellet composition, and subcritical neutron multiplication is performed automatically by MCNP. The Type 109 material composition is used in the MCNP models because it is the most reactive and maximizes the neutron dose rate due to enhanced subcritical neutron multiplication. However, the bounding Type 219 gamma and neutron sources are used in the MCNP models. Note that if the Type 219 source were modeled with the Type 219 pellet material composition, the dose rates would be less than the dose rates currently reported in the SAR.

In the text cited in the RAI, the text on page 5.2-6 is referring to the MCNP material composition, while the text on page 5.2-7 is referring to the source magnitude (gammas/s or neutrons/s).

The calculated cask surface dose rate for each of the TRIGA elements considered is provided in Table 5.2-8. These dose rates are based on MCNP models that feature the actual source term for the fuel type listed, although all models conservatively use Type 109 pellet material compositions. These dose rates are used to rank the various TRIGA element types. The maximum dose rate occurs for Type 219, with a cask surface dose rate of 61.6 mrem/hr. Type 219 bounds all other TRIGA element types for dose rate by a large margin. The cask surface dose rate of Type 109 is only 12.3 mrem/hr, which is significantly less. Type 219 is the bounding TRIGA element source term for all enrichments, burnups, and cooling times requested in this licensing amendment.

Sh-5-4. Clarify why the loose plates remain in the loose plate box under hypothetical accident conditions.

The applicant states the following in its application:

> "The loose plate box is not relied upon to maintain its configuration in the hypothetical accident, although the plates would remain within the basket compartments. It is assumed in the shielding models that the loose plates remain within the loose plate box in an accident because the bulk shielding provided by the box would remain largely unchanged if the box is damaged."

The applicant should demonstrate that loose plates stay in the loose plate box under hypothetical accident conditions, and that this demonstration is valid even if the box is damaged.

This information is necessary to determine compliance with 10 CFR 71.47(a) and (b).

Response: As discussed in the response to RAI G-1, the analysis of the loose plate box has been revised to show that the box does not deform under NCT or HAC, and the possibility for movement of the fuel plates will be restricted. Loose fuel plates will remain within the loose plate box under all transport conditions. Refer to the G-1 response for more details. Also, because the material of the loose plate box has been changed from aluminum to stainless steel, the MCNP shielding evaluations have been modified and rerun.

Additional Changes:

1. Drawing 1910-01-03-SAR, General Note 5, has been revised to remove aluminum welds from, and to include intermittent welds in, the weld types which are excepted from the liquid penetrant inspection requirement. Aluminum is no longer used as a material. Intermittent welds should not be inspected by liquid penetrant means since the start and stop of the welds may show a false indication reading.
2. General Note 19 has been added to drawing 1910-01-03-SAR to permit markings to be placed on any of the baskets to aid in verifying proper placement of fuel elements during loading. These markings have no effect on any safety evaluation.
3. Drawing 1910-01-03-SAR has been revised to show a changed feature for lifting the loose plate box. The prior design included two holes in two opposite side plates of the box. It was concluded that these holes might be difficult to use if the box is filled with material. The new lifting feature consists of trunnions, located essentially where the prior holes were located. The trunnions will be used with a small lifting yoke in a licensed facility. This change has no effect on any safety evaluation.
4. The PULSTAR gamma and neutron source terms and decay heat have been recomputed using a larger reactor power to reflect an upcoming power uprate at the PULSTAR reactor.