July 27, 2007

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON TS125 AMENDMENT REQUEST

Reference: Letter from Steven Sisley to U.S. NRC, dated October 1, 2006, Subject: Submittal of Request to Amend the FuelSolutions TS125 Transportation System CoC, NRC Docket No. 71-9276

Dear Mr. Sisley:

By letter dated October 1, 2006, FuelSolutions submitted an application for an amendment to Certificate of Compliance No. 71-9276, Revision 2, for the TS125 transportation package. In its application, EnergySolutions has requested the transport of 48 of the currently loaded VSC-24 Multi-assembly Sealed Baskets (MSBs) in the TS125 transportation packaging. In addition, EnergySolutions has requested evaluation of the the TS125 transportation package for “-96" compliant.

In connection with the staff’s review, we need the information identified in the enclosure to this letter. In addition, the staff recommends a meeting to discuss the enclosed request. Please reference Docket No. 71-9276 and TAC No. L24034 in future correspondence related to this request. If you have any questions regarding this matter, you may contact me at 301-492-3338.

Sincerely,

/RA/

Meraj Rahimi, Senior Project Manager
Licensing Section
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No: 71-9276
TAC No: L24034
Enclosure: Request for Additional Information

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TS125/VSC-24 TRANSPORTATION SYSTEM
DOCKET NO. 71-9276
TAC NO. L24034
REQUEST FOR ADDITIONAL INFORMATION

By letter dated October 1, 2006, FuelSolutions submitted an application for an amendment to Certificate of Compliance No. 71-9276, Revision 2, for the TS125 transportation package. In its application, EnergySolutions has requested the transport of 48 of the currently loaded VSC-24 Multi-assembly Sealed Baskets (MSBs) in the the TS125 transportation packaging. In addition, EnergySolutions has requested evaluation of the TS125 transportation package for “-96" compliant.

This request for additional information (RAI) identifies information needed by the U.S. Nuclear Commission (NRC) staff in connection with its review of the amendment. The requested information is listed by chapter number and the 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 to complete its review of the application and to determine whether that applicant has demonstrated compliance with the regulatory requirements.

Chapter 1 - General Information

1-1 Provide a table addressing the nineteen issues considered in the rulemaking process that resulted in the revised rule.

Those sections in the SAR that address the “-96" requirements should be identified in Chapter 1 of the TS125 SAR in order to assure that all the “-96" requirements have been addressed.

This information is needed to determine compliance with 10 CFR 71.31.

1-2 Identify the revision to the TS125 SAR regarding the change to 10 CFR Part 71 for “-96" with respect to double containment of plutonium.

Table 1.0-1 of the TS125 SAR identifies Section 4.4 of TS125 SAR addressing 10 CFR 71.63 requirements. However, there are no such sections in the TS125 SAR.

This information is needed to determine compliance with the requirements in 10 CFR 71.63.

1-3 Provide a general description of the contents for VSC-24 MSBs.

Chapter 1 of the VSC-24 SAR needs to contain general information with respect to the assembly types, range of cooling time, burnup, decay heat, initial enrichment, and Criticality Safety Index.

This information is needed to determine compliance with 10 CFR 71.33 (See Reg. Guide 7.9).

1-4 Describe and provide the reasons for all additions, deletions, or revisions in each of the drawing sheets in the TS125 SAR.

No descriptions or justifications are provided for the changes made in the certification drawings.

This information is needed to determine compliance with 10 CFR 71.33 .

Chapter 2 - Structural

Section 2.1.2.2

2-1 Provide an evaluation of the MSB Storage Sleeve Assembly in accordance with the provisions of the ASME B&PV Code, Section III, Subsection NG.

The SAR states “The MSB assembly is designed in accordance with the requirements for Class 2 components from Subsection NC of the ASME Code.” Subsection NC provides rules for the strength and pressure integrity of items whose failure would violate the pressure retaining boundary. The MSB Storage Sleeve Assembly has not been designed and fabricated to be a pressure retaining boundary, and therefore the rules of NC do not apply. Subsection NG provides rules for core support structures which are designed to provide direct support or restraint of nuclear fuel. The MSB Storage Sleeve Assembly provides direct support and restraint of spent nuclear fuel during transport, and therefore MUST be evaluated to the rules of Subsection NG.

This information is needed to confirm that the package meets the requirements of 10 CFR 71.31(c), 71.35(a).

2-2 Provide justification and documentation for the weld quality factors that will be used in the evaluation of all MSB Storage Sleeve Assembly welds in accordance with ASME B&PV Code Section III, Subsection NG.

Subsection NG, Table NG-3352-1 provides weld quality factors (n). The weld quality factor is used to multiply the allowable stress limit by the quality factor in evaluating the design margin. The weld quality factor for a specific weld depends on the type of weld and the level of NDE that was provided.

This information is needed to confirm that the package meets the requirements of 10 CFR 71.31(c), 71.35(a).

2-3 Provide an ANSYS analysis of the MSB Storage Sleeve Assembly with all interior nodes at the sleeve-to-sleeve connections coupled in ALL six degrees of freedom (i.e., UX, UY, UZ, ROTX, ROTY and ROTZ), not just the translational degrees of freedom (i.e., UX, UY and UZ).

Section 5.2.1.1 of Calculation Package No. VSC-03.3204 states "The partial penetration sleeve-to-sleeve..... welds are modeled as pinned connections, taking no credit for moment transfer through the welds." Modeling the sleeve-to-sleeve partial penetration weld connections as pinned connections results in a gross underestimate of the stresses in these welds and is unacceptable. The behavior of this sleeve-to-sleeve connection is complicated. The resultant moment occurring at this connection can essentially be separated into three parts: the moment components resisted by the two sleeves independently, the moment component resisted by the upper partial penetration weld in pure bending, and the moment component resisted by the lower partial penetration weld acting as the tension component of a force couple where the compression component of the couple is created by sleeve-to-sleeve contact. To understand this behavior the staff suggests that a detailed finite element model of this connection be developed using solid elements. In this way the moment calculated in the revised ANSYS model can be properly distributed to the structural components that makeup the sleeve-to-sleeve connection.

This information is needed to confirm that the package meets the requirements of 10 CFR 71.31(c), 71.35(a).

Section 2.2.2

2-4 Provide data or evidence to show that the amount of hydrogen gas generated inside the cask cavity or MSB assembly during on-site storage and preparation for transportation is less than the regulatory limit of 5% of the free gas volume.

During on-site storage and preparation for transportation, hydrogen gas may generate from radioactive decay inside the MSB assembly or cask cavity after loading operations. Should the amount exceed the critical limit, explosive concentrations of combustible gas may result. Assurance is needed by providing data to show that the amount of hydrogen gas produced is less than 5% of the free gas volume specified by the regulation. There is no Section 7.1.4 in the VSC-24 SAR, as referenced in the TS125 SAR, which is supposed to specify the vacuum drying time.

This information is needed to confirm that the package meets the requirements of 10 CFR 71.43(d).

2-5 Discuss how engineering non-conformances, NRC inspection findings, and other fabrication issues associated with the VSC-24 were considered in the design ana analyses. Justify adequate structural performance for VSC-24 canisters with any pre-existing weld cracks in the canister body and fuel sleeves.

Many fabrication and QA problems were identified for the MSBs in the 1990s. Examples include three rejectable flaws detected in a MSB vertical shell weld at Palisades, and shield lid weld cracking detected at Point Beach and Arkansas Nuclear One (ANO). The defects were analyzed and found to be acceptable for storage purposes, although delayed failure was a concern. However, given the conditions that there were pre-existing cracks present in the canister's vertical shell welds and shield lid welds, assurance is hereby requested, either thru testing or analysis, that the MSB payload assembly can still have adequate structural capability to withstand the mechanical loads during transport under NCT and HAC conditions.

This information is needed to confirm that the package meets the acceptance requirements of 10 CFR 71.31(c), 71.37(b).

Section 2.6.7

2-6 Provide data to show residual stresses produced by fabrication of the MSB payload assembly are insignificant.

The application states that internal pressure and fabrication stresses are insignificant; only thermal gradients and thermal expansion mismatch between dissimilar materials are important. Data are needed to show that residual stresses produced by fabrication of the MSB payload assembly are insignificant. (Please note: (1) the unit of weld shear stress in the last column of Table 6.2-4 on page 112 of 258 in the Calculation Package No. VSC-03.3204 should be **ksi**, not **psi**; (2) the unit of bending moment on page 17 of 28 in MSB Axial Spacer Structural Evaluation should be **in-kips per inch**, not **in-kips** for M_{ra} and **in-lb** for M_{rb}.)

This information is needed to confirm that the package meets the structural requirements of 10 CFR 71.31.

Section 2.6.7.1

2-7 Justify that the “gap closed” initial condition is the worst case scenario in the transient dynamic finite element analysis for the NCT and HAC side drop impact events.

In the NCT drop and the subsequent HAC drop analysis, the gaps between the cask internal surface, radial spacer and the MSB payload assembly are always set to zero as the initial condition for the transient dynamic finite element analysis. It is possible that other initial configurations such as the gaps being in the most open state between components may result in higher dynamic impact loads. Justification is needed to demonstrate that zero initial gaps give rise to the worst case scenario in terms of the magnitude of the resultant stresses in critical components.

This information is needed to confirm that the package meets the structural requirements of 10 CFR 71.31, 71.33, and 71.35(a).

Section 2.11

2-8 Provide a verification on the evaluations of post buckling fuel rod stresses under HAC end drop conditions.

Some of the derivations for the post buckling stress analysis of fuel rods result in an expression different from that obtained by the staff in confirmatory calculations. Examples include: peak deflection δ, stress S_{gsi} at the grid strap and maximum bending stress S_{bi} at the middle of unsupported section of the fuel rod. As a result, the critical g-load obtained significantly underestimates the true value.

This information is needed to confirm that the package meets the requirements of 10 CFR 71.7(a).

2-9 Provide high burnup fuel material's mechanical property data to show that the yield strength and Young's modulus are independent of the level of burnup.

The application claims that in the evaluation of high burnup fuel buckling behavior, the materials properties of low burnup fuels can be used as they retain sufficient strength and ductility. Mechanical property data on high burnup fuels are needed to justify this statement.

This information is needed to confirm that the package meets the requirements of 10 CFR 71.43(d).

Chapter 3 - Thermal

3-1 Justify the existence of convection in the larger gas regions and quantify the contribution of convection heat transfer to the overall removal of heat from the package.

On page 3-23 of the VSC-24 SAR, the applicant states: "For narrow regions of any orientation, little or no convective heat transfer will occur, and only conduction through the helium filled void spaces is assumed. Larger gas volume regions experience a significant level of convective heat transfer as illustrated in Figure 3.3-5."

Natural convection in complex horizontal basket designs (such as seen in Figure 3.3-5) should be validated through CFD calculations or physical experiments.

This information is needed to determine compliance with 10 CFR 71.7(a).

3-2 Explain why the highest temperature for the HAC event were less than the maximum for the NCT. Specify calculated temperatures using consistent units.

On page 3-6 of the VSC-24 SAR, the highest temperature for the HAC event were less than the maximum for the NCT.

This information is needed to determine compliance with 10 CFR 71.7(a).

3-3 Provide the validation or benchmarking process applicable to TS125 casks and canisters directly or by referencing calculation packages in which validation or benchmarking is done.

On page 3-21 of the VSC-24 SAR, it is stated that programs (SINDA/FLUINT) are "benchmarked and widely used for thermal analysis," however, the application does not specify of any validation or benchmarking specifically on casks. In the TS-125 SAR (page 3.6-8) it states that "thermal models of the FuelSolutions storage cask and canisters have been benchmarked against models developed for the FLUENT and COBRA-SFS codes and found to provide similar results." However, it does not appear a formal validation process was described for the SINDA/FLUINT code. Comparison to confirmatory analysis models done by the NRC in other licensing actions does not constitute an adequate benchmarking process.

This information is needed to determine compliance with 10 CFR 71.7(a).

3-4 Confirm any postulated convective flow in the VSC-24 Canister using a computational fluid dynamics (CFD) method. Explain exactly how the enhanced helium conductivity is applied in the analysis model.

The staff's expectation is that CFD analyses (or similar) should be used for confirmation of convective flow internal to the canister. On page 3-23 of the VSC-24 SAR the description: "helium gas conductivity values are increased by a factor of between 4 and 7 to account for the presence of convective flow..." is generally lacking in terms of where it is applied and what values are actually used for helium conductivity.

This information is needed to determine compliance with 10 CFR 71.7(a).

3-5 Provide the supporting documentation for the allowable temperature and exposure duration for the O-ring seals.

On page 3-25 of the VSC-24 SAR, it is indicated that the cask closure metallic O-ring seals are shown to have an allowable temperature of 932°F. Provide documentation that demonstrates performance of the seals to the temperature limit stated. The manufacturer's product data sheets should be included (or cited) to verify the allowable temperature and exposure duration for seal materials .

This information is needed to determine compliance with 10 CFR 71.7(a).

Chapter 4 - Containment

No additional information is needed.

Chapter 5 - Shielding

5-1 Explain the apparent discrepancies between the tabulated values for Assembly Thermal Power and the values used in the calculations.

Table 5.2-1 in the VSC-24 Transportation SAR indicates the Assembly Thermal Power assumed is 40.223 MWt/assembly. This agrees with the value in Table 2-1 in the referenced calculation (CMPC.1703.002). However, these tables do not seem to agree with the input data specified on page B5 of this calculation, or the example output on page C5 of this calculation. Both of these locations indicate 40.223 MWt is the power associated with one MTU, not one assembly.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC.

5-2 Generate and provide neutron and gamma source terms for each fuel type in this amendment (B&W 15×15, CE 16×16, CE 15×15, and W 14×14) accounting for differences in lattice configurations. Provide the ANO-1 (B&W 15×15) input and output files as an example.

Staff has not been able to confirm that scaling W 17×17 source terms for other assembly types is appropriate.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC.

5-3 Clarify Table 5.2-3 of the VSC-24 transport SAR to specify the gamma energy groups used to perform the shielding calculations. If any rebinning of the gamma spectrum was performed, discuss how it was performed.

The application is not clear on whether the photon energies in the first column of Table 5.2-3 in the VSC-24 SAR are group midpoints or endpoints or if the gammas were rebinned into these energy groups.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC.

5-4 Revise Section 5.3.1.3 to clarify all the changes between the NCT and HAC cases. Provide an example input file for the shielding evaluation for the bottom end HAC neutron model. If necessary, correct and re-run any affected shielding models.

Text on page 5-36, Section 5.3.1.3, in the VSC-24 Transportation SAR and page 18 of 153 in Calculation Package No. VSC-03.3501 indicates that credit is taken for the damaged (all H, O, and ^{10}B removed) NS-4FR neutron shield material in the bottom end gamma HAC models for the ANO MSBs. The FINITE SLAB BOTTOM END HAC GAMMA MODEL - ANO-2 NON-FUEL HARD SOURCE input deck on page 5-110 has a definition for “Dry NS-4FR Neutron Shield Material,” which does not appear to have had the ^{10}B removed from it. There is no mention of RX-277 in Section 5.3.1.3, but the Point Beach gamma HAC MCNP file that starts on page 5-104 of the VSC-24 SAR has a modified material composition for RX-277.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC.

5-5 Clarify the discrepancy between Table 5.3-1 and Drawings 5.3-18 and 5.3-20 regarding the type of cask bottom neutron shield material. If needed, correct and re-run any affected shielding models. Also, revise note 1 to Table 5.3-1 to indicate the honeycomb material is only modeled for NCT (not HAC).

Table 5.3-1 in the VSC-24 Transportation SAR indicates the cask bottom neutron shield consists of NS-4FR Neutron Shield Material. Drawings 5.3-18 and 5.3-20 show no NS-4FR, but do show RX-277 in its place. The same drawings appear in calculation package VSC-03.3501.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC.

5-6 Revise the discussion in the fourth paragraph of Section 5.3.1.4 (page 5-37) of the VSC-24 SAR so that the relationship between the locations of the dose points assumed for calculations and the actual distance from the conveyance that will exist at the time of shipment is clear.

This paragraph describes the two-meter surface used for assessing the HAC dose rates. From this discussion the location of the two-meter surface relative to the package, the actual conveyance, and the assumed conveyance is not clear.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC and 10 CFR 71.47(b)(3) requires that the dose rate 2 meters from the sides, front, or back of the vehicle do not exceed 0.1 mSV/h (10 mrem/h).

5-7 Correct the discrepancy between the TS125 SAR and the VSC-24 shielding calculation regarding the modeling of the bottom neutron shield in accident conditions, and if needed, re-run any affected shielding models.

Section 5.3.2, page 5.3-6, of the TS125 Transportation Cask SAR states "... For simplicity, the bottom end neutron shield is modeled as void for the accident condition." This contradicts information contained elsewhere (specifically in Calculation Package VSC-03.3501) for the VSC-24 MSBs containing ANO fuel.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC.

5-8 Clarify the nitrogen atom density for the dry neutron shield region mixture for the Point Beach HAC model run and re-run the analysis if needed. Correct any affected tables as appropriate.

The atom density for nitrogen in material 7 on page 5-107 in the VSC-24 SAR is an order of magnitude lower than that specified in Table 4-3 in Calculation Package VSC-03.3501 (page 33) and used in the other analyses.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC.

5-9 Revise Table 5.5-1 to specify the correct conversion factors for dose calculations. In addition, clarify the actual values used in the dose calculations.

In Section 5.5.1 of the TS125 SAR, page 5.5-2, Table 5.5-1, the ANSI/ANS-6.1.1-1977 flux-to-dose-rate conversion factor for a gamma energy of 0.50 MeV is listed as 1.15E-06. It should be 1.17E-6. This incorrect value also appears in the MCNP input files provided.

10 CFR 71.7 requires completeness and accuracy of all information provided to the NRC.

5-10 Clarify the basis for physical properties of RX-277 assumed in the shielding analyses, including any tests or analyses that account for potential aging effects from thermal or degradation. Explain how any changes in the properties such as density are considered in the shielding analyses.

It appears RX-277 in the MSB could have been subjected to potential heat and radiation degradation effects during long term storage in the VSC-24.

This information is needed to confirm compliance with 71.33.

5-11 Correct the following editorial errors.

- On the title page of the VSC-24 SAR, the Document number shown is 0000-0632. In the document headers the Document number is shown as 0000-0623. Please correct whichever of these is in error.

- In Table 5.2-7 on page 5-25 of the VSC-24 SAR, the “gammas/sec-MTU” in the second line of the title should be “neutrons/sec-MTU.”

- On page 5-71 in Section 5.4.1.4 of the VSC-24 SAR, there are two references to the “W125" Transportation Cask SAR. It appears that this should be the TS125 SAR.

Chapter 6 - Criticality

This following information is needed to confirm that the package meets the requirements of 10 CFR 71.31.

Section 6.2.1.1

6-1 Provide the irradiation history of the fuel pins used in the reconstituted Palisades fuel assemblies.

Section 6.2.1.1 references reconstituted assemblies such as “L2S” and “L3S.” Section 6.3.1.2.2.1 discusses these assemblies further. However, it is not clear where the depletion information for the parent assemblies from which fuel pins were used for the reconstituted assemblies are presented.

6-2 Clarify the origin of the fuel density values used in the depletion calculations.

It is not clear if the nominal or maximum values for fuel densities were used.

6-3 Justify the use of a more limiting one dimensional depletion computer code instead of a two dimensional computer code.

As indicated in the last paragraph of Section 6.2.1.1., the limitations of the one dimensional depletion computer code used through SAS2H module of SCALE does not allow modeling the variation in fuel enrichment. This is an important parameter which has direct effect on the reactivity through the discharged spent fuel inventory. In addition, using an average instead of maximum assembly enrichment is not bounding.

Section 6.2.2.4

6-4 Provide justification for the applicability of the recommended bounding axial profiles (that are not based on any data from W 14x14 assemblies) for the W 14x14 assembly analyses.

Recommended bounding axial burnup profiles from DOE/RW-0472 were used in the analyses. These recommended profiles were selected from a database of axial profiles that did not include any W 14x14 assemblies.

6-5 Justify the use of actinide-only axial bounding burnup profiles in DOE/RW-0472 for the actinide-plus- fission product burnup credit VSC-24 canisters.

The three-group bounding axial burnup profiles developed in DOW/RW-0472 was developed based on actinide-only credit. The same groupings and the same profiles may not be bounding for actinides-plus-fission product approach.

Section 6.2.1.2.2

6-6 Justify modeling a heterogenous fuel assembly geometry with a one dimensional homogenous model.

An ANO-1 assembly could have been exposed to nine different types of insertions during irradiation. This has a large impact on the neutron spectrum within the fuel assembly especially when inserts such as control rods and axial power shaping rods are modeled. Modeling the entire assembly with a one-dimensional computer code such as SAS2H does not account for the heterogeneity in the fuel assemblies.

Section 6.2.2.5.4

6-7 Justify your 25% assumption when calculating the cladding temperature.

It is unclear why the temperature difference between the water and the cladding was assumed to be 25% of the overall difference in temperature between the water and the fuel.

Section 6.2.2.6.3.1

6-8 Indicate which cases you are referring to in the second paragraph.

This paragraph starts out referring to Case 2, however the analyses that were performed refer to cases 3A and 3B.

Section 6.2.4

6-9 Justify the use of assembly-averaged isotopic compositions from the donor assembly for the individual fuel rods that have been moved to “receiver” assemblies.

It appears as though assembly-averaged isotopic compositions from the donor (or parent) assembly are used for the individual fuel rods that have been moved to “receiver” assemblies. It may not be appropriate to use assembly average compositions for rods from the donor assembly, particularly since many of these rods resided in peripheral locations in the donor assemblies and could have lower than average burnup.

Section 6.2.5.1

6-10 Provide justification for the applicability of the data from DOE/RW-0496 report to the CE and W 14x14 assemblies.

The reference source for horizontal burnup gradient data (DOE/RW-0496) does not include any data from CE plants/fuel or Westinghouse 2-Loop plants with W14x14 fuel.

Section 6.3.1.1

6-11 Justify using pure water instead of a mixture of water and steel for top and bottom nozzles are conservative.

At the beginning of the first full paragraph on page 6-90, it is indicated that pure water is used to model to and bottom nozzles. That may not be conservative.

6-12 Demonstrate that the use of nominal measurements for axial dimensions and exterior components will have no measurable effect on the reactivity of the system as opposed to using conservative worst-case fabrication tolerances.

There is no calculational analysis supporting this assumption.

Section 6.3.1.2.1

6-13 Provide sensitivity analyses that demonstrate using an average fuel zone height is conservative when modeling ANO assemblies.

This assumption may not be conservative.

Section 6.3.1.2.2

6-14 Justify that modeling the hafnium material in the I1h assembly inserts is a conservative assumption.

Other absorber rods are modeled as inserted into the assembly guide tubes but not the hafnium.

Section 6.3.3.2

6-15 Justify why several different cross-section library sets were used in the MCNP5 calculations.

Although this is probably to account for updated cross-section sets, no explanation was provided as to why these various sets were chosen.

Section 6.8.1.1

6-16 Justify the use of CRC models, including the isotopic compositions, that were developed by another organization.

Code bias and bias uncertainty are typically determined based on calculations for well-known, usually critical, systems and are intended to be representative of the modeling approximations, nuclear data, and calculational method used to model an application, e.g., storage cask.

Section 6.8.1.2

6-17 Justify the applicability and use of the Commercial Reactor Critical configurations for validation of the criticality analyses of the MSBs.

Isotopes such as Am-241 and Gd-155 build in with time, and have a significant negative reactivity worth at the long cooling times credited in this application. These isotopes are not present in any of the critical experiments that were analyzed and may not be present in appreciable quantities in the CRCs.

6-18 Describe data uncertainties associated with CRC models.

Given that the CRCs are complex systems that involve considerable input data in terms of material dimensions, compositions, reactor operating history information, and depletion isotopic compositions, a discussion on the data uncertainties in these configurations needs to be provided.

Section 6.8.2.1

6-19 Provide justification for using average modeling parameters for depletion validation calculations.

Towards the end of the section, the statement that the use of average modeling parameters for benchmark depletion calculations will lead to random errors and larger uncertainty is not supported. It would seem more likely to result in a systematic bias. Also, this is inconsistent with the level of detail used for the licensing basis analyses.

Section 6.8.2.2

6-20 Justify the use of the code calculations to obtain measured data.

The measured concentrations in the SAR are not consistent with the published measurements. The discussion in Section 6.8.2.2 states that the PNNL data for combined parent-daughter isotopes were used and split into individual isotopes according to the calculated concentrations (e.g., when Sm-147+Pm-147 were reported as combined concentrations, the separate Sm-147 and Pm-147 concentrations were determined by splitting the reported total using calculated concentrations). Hence the code to be validated (SAS2H) is being used to derive measured data, which is not justified.

Section 6.8.2.3

6-21 Describe how the isotopic benchmarks cover the ANO-1 spent fuels.

ANO-1 has nine different inserts that have been used during irradiation. They include control rods, power shaping rods, poison rods, orifice rods, etc. How are the effects of these rods on the isotopic inventories benchmarked? Provide additional justification for validation coverage.

The validation coverage of the loaded fuel is not conclusive. Although it is stated that the burnup range is covered, the evaluation is based on comparing rod segment burnup values (validation) with assembly average burnup values (application). The low burnup regions at the fuel ends, that are considerably lower than the assembly average burnup, drive reactivity and are not well represented by the validation data. According to Table 6.8-15, even the rod segment rod minimum burnup for fission product benchmarks does not cover the minimum burnup value for the loaded MSBs.

Section 6.8.2.4

6-22 Provide normality tests and trending analyses for the measured/calculated isotopics.

As part of developing the correction factors, the measured/calculated ratios have to be first tested for normality. Subsequently, a trending analyses with respect to parameters listed in Table 6.8-15 (i.e., rod pitch, water-to-fuel ratio, U-235 enrichment, burnup, cooling time, and AENCF) needs to be performed. For isotopes such as Mo-95, Ru-101, Rh-103, Ag-109, and Cs-133 with only three data points, small-sample or non-normal statistics, depending on the normality test, needs to be used.

Section 6.8.2.5.3

6-23 Develop biases for Am-241 based on Am-241 data instead of Pu-241.

The argument that the measurement data for Am-241 is sparse is not justified. There are data available from more than 40 samples for Am-241.

6-24 Explain why the missing experimental data for U-238 is not treated in a manner consistent with other missing nuclides and determine the impact on the analysis if the isotopic uncertainty for these samples is included.

The treatment of samples with missing U-238 measurements appears to be inconsistent with other isotopes with missing measurements. The calculated concentrations appear to be used in place of measured concentrations without any adjustment for uncertainty, implying no calculational uncertainty for U-238. Even though the calculated uncertainty is acknowledged to be relatively small, U-238 is one of the largest absorbers and can make a significant contribution to the overall delta-k uncertainty.

Section 6.8.2.5.5

6-25 Clarify the statement: "The issue of Δk_{eff} distribution normality (discussed in Section 6.8.2.5.4) is not applicable to the USL method."

The base assumption for USL is normal distribution. In addition, the discussion presented in Section 6.8.2.5.4 for 95% confidence using two sigma is based on normality assumption. Therefore, it is not clear why the applicant believes normality assumption does not apply to upper safety limit (USL).

6-26 Present the values in Table 6.8-23 and 6.8-25 vs. the values in Table 6.8-24 graphically.

The SAR states that a USL statistical analysis was performed for six fuel and depletion parameters. The results are presented in Tables 6.8-23, 24, 25, and 26. However it is difficult to see if the trending results make any sense without showing a plot of the data. Illustrate these results and indicate the degree of correlation in the trend data and whether the trends are statistically significant.

6-27 Explain why the isotopic measurement uncertainties were not included in the total uncertainty.

In the USL method, it is stated that the delta-k_{eff} uncertainty used in the analysis is due to the MCNP statistical error. The uncertainties in the experimental isotopic measurements do not appear to have been included, and would most likely be much larger than code stochastic error. What is the impact of the omission of measurement errors on the USL analysis?

Section 6.8.2.5.5

6-28 Demonstrate that the trends with the fuel parameters are not determined by the difference in the number of nuclides with actual measured data available in each sample.

The combination of actual measurements with supplemented bounding calculated concentrations raises a question about whether the USL trending analysis performed has any basis. The separate analysis of actinide and fission product uncertainty is performed to reduce the effect of mixing data sets with different isotopic measurements, but even with this approach the delta-k values are obtained using sets with different measured isotopes. This raises the question of whether the trending analysis performed using the fuel parameters is related to the parameters or to the different nuclides measured in the different samples – particularly for the fission products.

Section 6.9.2

6-29 Specify the risks associated with lowering the criticality margin by 60% (i.e., k_{eff} = 0.98) versus unloading the MSBs and reloading them into a different waste transportation packages that meets the standard criticality margin of 0.95. Provide a basis for the risk methodology considered in the comparison of the two alternatives.

For the two alternatives that are compared, the basis should specify the hazards and accident scenarios, exposed populations, assumed consequences (e.g., normal and accident doses), the appropriate risk acceptance criteria (or risk metrics), and/or other factors such as safety margins.

Section 6.9.2.1

6-30 Provide the detailed information on how the burnup values were calculated and measured including the associated uncertainties.

Present the documentation on the methodology and the analyses on arriving at the calculated burnup values in the reactor records. Describe the measurements in details and the parameter used in deriving to the measured burnups. Provide the standard deviation of each of the two methods employed in determining the assembly burnup for the 184 ANO-1 assemblies. Compare the combined standard deviation for the two methods,

i.e., $\sigma = \text{sqrt}(\sigma^2_R + \sigma^2_M)$,

where subscripts *R* and *M* refer to 'recorded' and 'measured' values, respectively, to the standard deviation of the measured/recorded burnup ratio distribution (0.03946). Provide the basis for the assembly burnup values used throughout the SAR analysis. If recorded burnup values were used, provide a discussion on how the uncertainty was factored into depletion/criticality analyses, including the impact of assembly burnup uncertainties on the k_{eff} values for the MSBs with small uncertainty margins (e.g., MSBs #2 & #15).

6-31 Provide justification as to why a normal distribution is appropriate when discussing the burnup ratio distribution.

No testing for normality was included in the SAR. This needs to be justified since the under-burn calculations are based on this assumption.

Section 6.9.2.2

6-32 Describe the basis for assigning the probability of loading individual under-burned spent fuel assembly.

It's not clear if the total probability of 1×10^{-6} is the starting point and the probability of individual under-burned assembly is derived from this number. If that is the case the basis for the total probability is not clear.

Section 6.9.2.3

6-33 Justify why MSB #22 can be used as the representative case for the second group when MSB #18 of that same group has a lower safety margin.

Since the applicant is using bounding representative cases for each group, it would appear that this is a non-conservative case to be used as the example.

6-34 Justify why each of the seven MSBs that exceed the accepted margin of subcriticality (i.e., 0.95) were not modeled independently and were placed into groups instead.

Since such a small set of MSBs exceed the typical margin of subcriticality, it is not clear why each of the seven cases were not run independently. As the margin is reduced greater confidence in the accuracy of the cases modeled is needed.

Section 6.9.2.5

6-35 Explain and justify why for each set of fuel parameters two cases are studied, one nominal case, and one case where the burnup level is reduced by 10% for all 24 assemblies in the MSB.

It is unclear why 10% was chosen for the burnup level reduction.

6-36 Describe how well the sample of spent fuel assemblies chosen from ANO-1 for burnup measurements represent the integrity of the burnup records for the remaining spent fuel assemblies from Palisades and Point Beach loaded into the VSC-24 canisters.

As indicated in the Interim Staff Guidance-8, Rev. 2, "Procedures that confirm the reactor records using measurement of a sampling of the fuel assemblies will be considered if a database of measured data is provided to justify the adequacy of the procedures in comparison to procedures that measures each assembly." It is not clear how the burnup measurements performed on a sample of spent fuel assemblies from ANO-1 verifies the Palisades and Point Beach's records on burnup for spent fuel assemblies discharged from those sites.