October 14, 2009

Mr. Richard Boyle, Chief
Radioactive Materials Branch
Office of Hazardous Materials
Technology
U.S. Department of Transportation
400 Seventh Street, S.W.
Washington, DC 20590

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE MODEL NO. NCS 45 PACKAGE

Dear Mr. Boyle:

This refers to your request dated January 22, 2009, for a recommendation concerning the revalidation of the Model No. NCS 45 package, German Certificate of Approval No. D/4347/B(U)F-96, Revision 0. Enclosed are requests for additional information needed to continue the review for revalidation of the NCS 45 package. Our established schedule provides a Certificate of Compliance issuance date of January 6, 2010, for the NCS 45 package. We request that you provide this information by November 13, 2009, or earlier if possible. Inform us at your earliest convenience, but no later than October 27, 2009, if you are not able to provide the information by that date. To assist us in re-scheduling your review, you should include a new proposed submittal date and the reasons for the delay.

Please reference Docket No. 71-3084 in future correspondence related to this request. The staff is available to meet to discuss your proposed responses. If you have any questions regarding this matter, I may be contacted at (301) 492-3294 or you may contact Chris Staab of my staff at (301) 492-3321.

Sincerely,

/RA/

Steven Baggett, Acting Chief
Licensing Branch
Division of Spent Fuel Storage
and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-3084
TAC No. L24301

Enclosure: Request for Additional Information

October 14, 2009

Mr. Richard Boyle, Chief
Radioactive Materials Branch
Office of Hazardous Materials
Technology
U.S. Department of Transportation
400 Seventh Street, S.W.
Washington, DC 20590

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF THE MODEL NO. NCS 45 PACKAGE

Dear Mr. Boyle:

This refers to your request dated January 22, 2009, for a recommendation concerning the revalidation of the Model No. NCS 45 package, German Certificate of Approval No. D/4347/B(U)F-96, Revision 0. Enclosed are requests for additional information needed to continue the review for revalidation of the NCS 45 package. Our established schedule provides a Certificate of Compliance issuance date of January 6, 2010, for the NCS 45 package. We request that you provide this information by November 13, 2009, or earlier if possible. Inform us at your earliest convenience, but no later than October 27, 2009, if you are not able to provide the information by that date. To assist us in re-scheduling your review, you should include a new proposed submittal date and the reasons for the delay.

Please reference Docket No. 71-3084 in future correspondence related to this request. The staff is available to meet to discuss your proposed responses. If you have any questions regarding this matter, I may be contacted at (301) 492-3294 or you may contact Chris Staab of my staff at (301) 492-3321.

Sincerely,
/RA/
Steven Baggett, Acting Chief
Licensing Branch
Division of Spent Fuel Storage
and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-3084 TAC No. L24301
Enclosure: Request for Additional Information
DISTRIBUTION: SFST r/f NMSS r/f BWhite, NMSS
G:\SFST\Staab\DOT Cases\NCS 45 RAI and SER\RAI Letter Two.doc **ML092880682**

OFC:	SFST	E	SFST		SFST		SFST		SFST		SFST	
NAME:	CStaab		JPiotter		NJordan		ZLi		CBajwa		MGordon	
DATE:	9/25/2009		10/9/2009		10/6/2009		9/25/2009		10/7/2009		10/9/2009	
OFC:	SFST		SFST		SFST		SFST		SFST		SFST	
NAME:	MWaters		CRegan		MRahimi		MDeBose		SBaggett			
DATE:	9/25/2009		10/13/2009		10/13/2009		10/14/2009		10/14/2009			

C = COVER E = COVER & ENCLOSURE N = NO COPY OFFICIAL RECORD COPY

Department of Transportation

Docket No. 71-3084

Request for Additional Information

Model No. NCS 45 Package

By letter dated January 22, 2009, the Department of Transportation submitted a request to the U.S. Nuclear Regulatory Commission to provide a recommendation concerning the revalidation of the Model No. NCS 45 package, German Certificate of Approval No. D/4347/B(U)F-96, Revision 0. This Request for Additional Information (RAI) identifies information needed by the U.S. Nuclear Regulatory Commission (NRC) staff in connection with its review of the application. Each individual RAI describes information needed by the staff for it to complete its review of the application to determine whether the applicant has demonstrated compliance with regulatory requirements.

Structural

1-1: Clarify the type of spruce wood used in the shock absorbers.

Section 3.6 of the Safety Analysis Report (SAR) states that the shock absorber is made of balsa and spruce wood, but according to the United States Department of Agriculture (USDA) Wood Handbook, there are multiple varieties of spruce, with varying mechanical properties.

In the response to a previous RAI, the applicant clarified the apparent discrepancy between the use of pine and spruce for the impact absorber, but the type of spruce used in the package will affect the mechanical properties of the wood. The staff is under the impression that the spruce chosen will have properties similar to those listed in NCS 0017, Rev. 4. This clarification is requested in combination with 1-2, as the mechanical properties of the woods used in the package, and used to demonstrate compliance with the regulations is unclear.

The Staff notes that the USDA Wood Handbook is available for download at:
http://www.fpl.fs.fed.us/products/publications/several_pubs.php?grouping_id=100&header_id=p

This information is required to meet TS-R-1 Regulation 656.

1-2: Justify the unusually high compressive strengths for the wood shock absorbers used in the structural analysis of the package (Table 2 of B-TA-3991-Rev. 2). If necessary, make corrections to the structural analysis of the package. Correct the apparent discrepancy between the compressive strengths for the wood shock absorbers in Table 2 of B-TA-3991-Rev. 2 and Table 4-30 of NCS 0017, Rev. 4.

According to the USDA Wood Handbook, the compressive strength of balsa and all types of spruce wood are significantly lower than the strengths quoted for these woods in Table 2 of B-TA-3991-Rev. 2. The staff agrees that the absorbed energy densities of the woods quoted in the application are appropriate, however.

This information is required to meet TS-R-1 Regulation 656.

1-3: Provide support for the “time yield limit” given for lead in Table 4-9 of SAR NCS 0017 Rev. 4. The response should include an estimate for the stress on the lead shielding during normal conditions of transport.

The tables in the Guruswamy reference do not support the values given in Table 4-9. These values are used in Sec. 4.2.4.3.1 to calculate heat effects. In response to a prior RAI, the applicant stated that the plot in Guruswamy was off by a factor of 100. The staff considers that the plot in Guruswamy is most likely off by a factor of at least 10,000 and requests clarification, and additional support for the time yield limit given for lead in Table 4-9 of SAR NCS 0017 Rev. 4.

This information is required to meet TS-R-1 Regulation 651(b).

1-4: Exclude the use of brazing to seal the cans intended to store radioactive material, or specify that the brazing material will have a melting temperature of greater than 411°C, the maximum center loading temperature under hypothetical accident conditions (HAC). The melting temperature of the brazes used should incorporate a margin of safety.

The relatively low melting temperature of brazes (compared to weld metal) could make brazes susceptible to melting during HAC.

This information is required to meet TS-R-1 Regulation 651(a).

1-5: Specify canning of high-burnup (> 45 GWd/MTU) spent nuclear fuel or provide justification for maintaining fuel configuration during hypothetical accident conditions (HAC) of transport.

ISG -11, Rev. 3, “Cladding Considerations for the Transportation and Storage of Spent Fuel, states:” “for high burnup cladding material, cladding performance during hypothetical accident conditions of transport will require further information on the impact properties.” The staff finds EPRI document 1009929 an insufficient basis for the mechanical properties of high-burnup spent nuclear fuel.

This information is required to meet TS-R-1 Regulation 651(a).

Reference:
“Spent-Fuel Transportation Applications: Modeling of Spent-Fuel Rod Transverse Tearing and Rod Breakage Resulting from Transportation Accidents.” Electric Power Research Institute. EPRI 1009929, October 2006.

1-6: Specify canning of aluminum-clad and stainless steel-clad spent nuclear fuel of different burnups or provide justification for maintaining fuel configuration during hypothetical accident conditions (HAC) of transport.

The properties of aluminum-clad and stainless steel-clad spent nuclear fuel are not as well documented as zirconium clad fuels. The staff notes that the properties of aluminum will be influenced by the anticipated decay heat.

This information is required to meet TS-R-1 Regulation 651(a).

Shielding

2-1: Provide justifications for the accuracy of or review and correct (if necessary) the source terms provided in Tables 7-37 and 7-41 of the SAR and update the shielding analysis (if necessary) based on updated source terms.

The applicant provides, in Tables 7-37 and 7-41 of the SAR, the source terms as a function of burnup for gamma and neutron radiation sources respectively. However, review of the data indicates that these values do not appear to be correct because they do not follow a well understood phenomenon that the gamma source is linearly proportional to the fuel burnup and the neutron source is proportional to the fourth power of fuel burnup [Ref. 1, 2], i.e.,

$$S(\gamma) \propto BU \quad (1)$$

$$S(n) \propto BU^4 \quad (2)$$

Where $S(\gamma)$ and S(n) are the gamma and neutron source strengths respectively and BU is the fuel assembly burnup. In addition, the staff's source term calculations for different PWR and BWR fuel assemblies at various burnups confirm the validity of the above relationships.

Because correct source terms is the basis of the shielding safety analysis, all radiation shielding analyses for the cask become void if the incorrect radiation source strength data were used. The applicant is requested to provide justification for the accuracy of the source terms as a function of burnup or correct the data and, if necessary, provide an updated shielding analysis based on the updated source term data.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-2: Provide justifications for not including the axial peaking of the source terms caused by burnup peaking in the middle section of the BWR or PWR rods in the source term calculations.

The Certificate of Approval indicates that PWR and BWR fuel rods are requested contents of the NCS 45 packages. However, it appears that the applicant did not include consideration of the axial peaking of the source term as a result of the uneven burnup along the axial direction of fuel assemblies. In general, the middle section of a fuel assembly incurs a much higher burnup due to flux peaking in the core. Typically, a burnup profile is used to adjust the source terms that were determined based on the average fuel assembly burnup using computer code such as SAS2H. NUREG/CR-6801 provides a large collection of the fuel profiles for various BWR and PWR fuel assembly designs of all major vendors.

The applicant used SAS2H in its source term calculations for the proposed contents but did not include considerations of the peaking of the source terms due to axial burnup peaking. As a result, the source term data might have been incorrectly calculated. The applicant is requested to provide justifications for not including the axial peaking of the source terms caused by burnup peaking in the middle section of the BWR or PWR rods in the source term calculation.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-3: Pertinent to the determination of the radiation source terms, the applicant is requested to provide:

1. details of the fuel assembly geometries, burnable poison content data (if any), specific power, fuel and moderator temperatures, moderator density, soluble boron (if applicable), and depletion histories;
2. justifications for using these data in the fuel depletion analysis for the determination of correct radiation source terms, including consideration of the differences between PWR and BWR fuels; and
3. update, if necessary, the source terms and the shielding safety analyses.

Fuel assembly geometries, burnable poison (if any) content data, specific power, fuel and moderator temperatures, moderator density, soluble boron (if applicable), and depletion histories are critical data for determining the source terms of the spent fuels. The applicant however did not provide any information in the SAR for these parameters. As a result, the staff is unable to recreate source term calculation models to verify the applicant's source term data presented in the SAR.

In addition, on page 7-70 of the SAR, the applicant states: "For longer delay times the influence of the irradiation history is no longer significant." Based on publications, this conclusion does not seem to be valid because lower moderator density and higher power density will increase the production of actinides, such as curium and californium, which are major neutron emitters in irradiated fuels. Studies published in NUREG/CR-6802 indicate that the neutron and secondary gamma increase as cooling time increases [Ref. 1]. The applicant is requested to provide: (1) details of the fuel assembly geometries, burnable poison content data (if any), specific power, fuel and moderator temperatures, moderator density, soluble boron (if applicable), and depletion histories; (2) justifications for using these data in the fuel depletion analysis for the determination of correct radiation source terms, including consideration of the differences between PWR and BWR fuels; and (3) update, if necessary, the source terms and the shielding safety analyses.

This information is needed for the staff to perform confirmatory analysis and determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-4: Pertinent to the solid nonfuel materials contents of the NCS 45 package:

1. provide detailed information about these materials;
2. explain how the radiation source terms from the solid nonfuel materials were calculated; and
3. explain how the radiation sources from these nonfuel materials were included in the shielding analysis for the NCS 45 package.

Table 1 of Attachment 1 of the Certificate of Approval indicates that solid nonfuel materials are acceptable contents of the NCS 45 packages. The shielding safety analysis however does not appear to have included these materials in the dose rate calculation equations, i.e., the source term data presented in Tables 7-37 and 7-41 do not seem to include nonfuel materials. Although the applicant states in its response to RAI-8 that stainless steel cladding is automatically included in source term calculations, the applicant did not provide information on how the source terms for nonfuel materials were calculated, nor the nature and compositions of

the nonfuel materials. The applicant is requested to provide detailed information on how the radiation source terms from the solid nonfuel materials were calculated and included in the shielding safety analysis for the NCS 45 package. The requested information include the source terms determination calculation and demonstration of inclusion of the source terms from the solid nonfuel materials in the shielding analysis.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-5: Pertinent to the adequacy of the design basis content, provide:

1. detailed information on the design basis contents of the R52 packages based on which the dose rates and source type correlations were established and used in the NCS 45 package dose rate calculations, and
2. justification for the adequacy of using the dose rates factors calculated based on the content of UO_2 compound with 4.5 w% U-235 enrichment, 60,000 MWd/tU burnup and 365 days of decay time as the design basis content to bound dose rate calculations for all contents of the NCS 45 packages.

In Section 7.3 of Chapter 7 of the NCS 45 SAR, the applicant described the calculation method and its verification and validation. From these descriptions, it seems that a set of formulas were developed based on the source terms and the corresponding measured dose rate data of the R52 packages. These formulas were then used in determining the dose rates at various locations of the NCS 45 package for given source terms (neutrons and gammas).

In addition, in Section 7.4, the SAR states that source terms were determined based on a fuel content of 4.5 w% U-235 in UO_2 compound with a 60,000 MWd/tU burnup and a 365 days of decay time. However, the SAR indicates, in Tables 7-36, 7-37 and 7-41, that the NCS 45 package is designed to transport contents with enrichment ranging from 1.0 w% to 5.3 w%, burnup ranging from 10 GWd/MgU to 120 GWd/MgU, and cooling time ranging from 120 days to 3650 days (10 years). Because of the vast variations in neutron and gamma spectra for these different contents, the design basis content may not be adequate for dose rate calculations for all of these contents. For example, low enrichment fuel will have stronger neutron source in comparison with that of the higher enrichment fuels and fuel with longer cooling time will have higher neutron contribution to the dose rates in comparison with fuel with short decay time. Gamma source will dominate for the contents with shorter cooling time.

The applicant is requested to provide: (1) detailed information on the design basis contents of the R52 packages based on which the dose rates and source type correlations were established and used in the NCS 45 package dose rate calculations, and (2) justification for the adequacy of using the dose rates factors calculated based on the content of UO_2 compound with 4.5 w% U-235 enrichment, 60,000 MWd/tU burnup and 365 days of decay time as the design basis content to bound dose rate calculations for all contents of the NCS 45 packages.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 2 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-6: Provide justification for the validity of the methodology developed and employed in the shielding analysis for the NCS 45 package design.

The shielding analysis chapter of the SAR describes the methodology and formulas developed based on R52 package dose rates versus source term ratios and used in the NCS 45 package shielding analysis. From the description of the methodology, it seems that the formulas developed based on the R52 package may not be valid because the gamma shielding layer is significantly thinner for NCS 45 in comparison with that of the R52 package. In addition, it seems that the methodology developed for the dose rate calculation does not provide reliable results. For example, from Table 7-5, it seems that for most of the cases, the proposed method underestimates the dose rates. For the worst case (case 5), the measured is 227 percents (2.27 times) of the calculated (378/166.6=2.27). The method developed by the applicant seems highly unreliable and cannot provide a reasonable assurance that the estimated doses are conservative. The applicant is requested to provide justification for the validity of the methodology developed and employed in the shielding analysis for the NCS 45 package design and, if necessary, a revised shielding analysis based on updated information.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 2 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-7: Provide justification for the use of the Origen/Arp code for burnup beyond its validated range.

Attachment 7-2 of the shielding analysis chapter of the NCS 45 SAR states: "Gamma and neutron source strengths are calculated with the program SAS2 [SAS2]. Basis of calculations are input files which were used for the calculation of ORIGEN-ARP [ORIGEN-ARP] libraries and which were validated there. Libraries contained in ORIGEN-ARP cover burn-up up to 70,500 MWd/MgU. In the following source strengths for burn-up up to 120,000 MWd/MgU are listed with no comprehensive validation exists." From these statements, it seems that the applicant is aware of the inadequacy of the ORIGEN-ARP code and its associated libraries. However, no further adjustments or penalty were applied to compensate this deficiency. As a result, the method developed by the applicant does not seem to be reliable and cannot provide a reasonable assurance that the estimated doses are conservative. The applicant is requested to provide justification for the use of the Origen/Arp code beyond its validated range and a revised shielding analysis, if necessary, for the NCS 45 package design based on an updated source term calculation.

This information is needed for the staff to perform confirmatory analysis and determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 2 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-8: Provide justification for the assumption that the cement layer will remain unchanged under Hypothetical Accident Condition.

In Section 7.2.4 of the shielding analysis chapter of the NCS 45 SAR, the applicant describes the assumptions used in the shielding safety analysis under hypothetical accident conditions. It seems from the shielding analysis model provided in Section 7.6 of the SAR that no crack in cement layer was assumed in the model. However, it seems very unlikely that the cement layer

will survive the tests prescribed in paras. 727 and 728 of TS-R-1 without resulting in cracks. The cracks in the cement layer will reduce the effectiveness of the neutron shielding of the package. As a result, the applicant's shielding analysis based on the stated assumption may not be valid. The applicant is requested to provide justification for the assumption that the cement layer will remain unchanged under Hypothetical Accident Condition and a revised shielding analysis, if necessary, for the NCS 45 package design based on updated assumption.

This information is needed for the staff to perform confirmatory analysis and determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 2 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-9: Provide justification for the use of different content definitions in different parts of the SAR for the NCS 45 package design.

The applicant indicates in the Certificate of Approval that PWR and BWR fuel rods are the contents of the packages and the characteristics of these fuel assemblies were used in the SAS2H models to determine the source terms for the contents. However, the criticality safety analysis chapter of the SAR indicates that the fuel pin geometry is 1 millimeter (1 mm) in diameter (page 8-16 of the SAR). This is not consistent with any of the current PWR or BWR fuel designs. The applicant is requested to provide justification for the use of different content definitions and clear and consistent definition for the contents.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 2 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-10: Correct the errors in the units for the data in Tables 7-2 and 7-3.

There appears to be some errors regarding the unit in the dose rate factors in Tables 7-2 and 7-3 in the shielding safety analysis of the NCS 45 SAR. The applicant is requested to review and correct, if necessary, the errors in the table.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 2 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-11: Demonstrate assertion that for the same total source strength the dose resulting from sources extending over the whole packaging length is smaller than for concentrated small source.

In its response to the staff's Request for Additional Information 4-2 (ML091820308), the applicant states that for the same total source strength the dose resulting from sources extending over the whole packaging length is smaller than for concentrated small source. The staff reviewed this assertion and found it can not come to the same conclusion. The staff's understanding is that this assertion may be acceptable only for certain special scenarios. The applicant needs to use a three dimensional transport theory based computer code to demonstrate this conclusion because it is in general not an acceptable assumption. For example, unless collimated source is assumed, an isotropic point source may provide smaller dose rate when the detector is close to the point source because the detector can cover only a small solid angle between the source and the detector. The applicant is requested to

demonstrate assertion that for the same total source strength the dose resulting from sources extending over the whole packaging length is smaller than that of a concentrated source.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 2 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-12: Provide information on how the source term from stainless steel components and cladding were automatically considered.

Confirm that the shielding design of the NCS 45 is not based on measured data.

In its response to the staff's Request for Additional Information 4-8 (ML091820308), the applicant states: "The gamma and neutron source term of the fuels to be transported must be known, measured or calculated, as appropriate. With the formulas it is checked that the dose rates are to be expected within the allowable limits." However, it is not clear what these statements mean. Does the applicant mean that the compliance of the package design relies on the actual dose rate measurements of the package? It is important to point out that the applicant must demonstrate by design that the package meets all regulatory requirements, including dose rates. The applicant is requested to confirm and assure that the shielding design of the NCS 45 is bounding for all authorized contents and does not rely on the measurement of the actual package dose rates.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 2 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-13: Provide justifications for the accuracy of or review and correct (if necessary) the decay heat provided in Table 5-2 of the Sar and update the thermal analysis (if necessary) based on updated decay heat.

The applicant provides, in Tables 5-2 of the SAR, the maximum allowed decay heat. The shielding analysis, Chapter 7, of the SAR indicates that the maximum payload is 100 Kg of PWR or BWR spent fuel with 120 GWd/MTU burnup and 120 days of cooling time. The staff's confirmatory analysis on the decay heat for the same amount of spent fuel materials shows that the maximum decay heat is equivalent to 100 Kg fuel with 60 GWd/MTU burnup and 120 day cooling time. The applicant is requested to provide justification on why the maximum of 3000 Watts decay heat provides the upper bound for the payload decay heat.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of paras. 652 and 653 of IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

2-14: Pertaining to gamma shielding safety evaluation:

1. Explain if lead slump was considered plausible during a Hypothetical Accident Conditions;
2. Provide justification why lead slump was not considered in the shielding evaluation;

3. Provide evaluation for shielding safety with adequate consideration of lead slump during hypothetical accident conditions.

The applicant describes the assumptions for accident conditions of transport in Section 7.2.4 of the SAR. However, it was not clear if lead slump was considered as a plausible damage to the gamma shielding layer.

Lead slump is a well understood phenomenon in package drop test. Unless demonstrated through test, lead slump is always considered in shielding analysis for accident condition. The applicant is requested to provide information on whether lead slump was considered plausible in accident conditions. If not, the applicant is requested to provide justification on why lead slump is not considered plausible.

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 531 and para. 532 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

Thermal

3-1: Provide justifications for the accuracy of or review and correct (if necessary) the decay heat provided in Table 5-2 of the SAR and update the thermal analysis (if necessary) based on updated decay heat.

Table 5-2 of the SAR states the maximum allowed decay heat is 3000 watts. Tables 6, 7, 8, 9, 10, and 11 of the SAR indicate that the maximum payload is 100 Kg of PWR or BWR spent fuel with 120 GWd/MTU burnup and a minimum 120 days of cooling time. The staff's confirmatory analysis on the decay heat for the same amount of spent fuel materials indicates that the maximum decay heat is equivalent to 100 Kg fuel with 60 GWd/MTU burnup and 120 day cooling time. The staff's analysis shows that the decay heat is about 4000 watts for 100 Kg fuel with 80 GWd/MTU burnup and 120 days cooling time. The applicant is requested to provide justification for why the maximum of 3000 Watts decay heat provides the upper bound decay heat for the proposed payloads (maximum 120 GWd/MTU, 120 days cooling time).

This information is needed for the staff to determine if the shielding design of the NCS 45 meets the requirements of para. 652 and para. 653 of the IAEA's Regulations for the Safe Transport of Radioactive Material TS-R-1.

Criticality

4-1: Provide justification for using SCALE 5 verification and validation (V&V) as the basis for SCALE 4.

The original RAI request was for inclusion of proper justification for the use of SCALE 5. This was based on the fact that a V&V reference was used which referred to SCALE 4 as opposed to SCALE 5. In addition, the response that was provided failed to indicate whether the "Guide to Verification and Validation of the SCALE-4 Criticality Safety Software" can be used for SCALE 5.

This information is required to meet TS-R-1 671(a), 677, and 679.

4-2: Provide the details used in the determination of the most reactive case used as the basis to analyze Content 1.2.

Based on the information given in the response to RAI 5-2 and the subsequent teleconference, the reviewer was able to re-calculate some of the results provided in Table 8-7. However, some of the water fractions yielded a 3 to 4% difference. To ensure that all assumptions used are fully understood please provide an example describing how the fissile material distribution, lattice spacing, and active diameter are used to determine the water fraction for the most reactive case. Please include example input files for cases involving the following fissile material distributions: 5.4156 g/cm, 26.2983 g/cm, and 50.0981 g/cm.

This information is required to meet TS-R-1 671(a), 677, and 679.

4-3: Provide more detail with input files used to determine the optimum fissile material distributions and water fractions.

Following the methodology used to calculate the bounding configuration for Content 1.3, the reviewer calculated different results for fissile material distributions and resulting water fractions. The calculated water fractions yielded approximately a 5.5% difference which could have merely been based on the minor difference in comparative fissile material distribution results. To ensure that all assumptions used are fully understood please provide an example describing how the fissile material distribution, lattice spacing, and active diameter are used to determine the water fraction for the most reactive case. Please provide example input files for three different fissile material variations at the bounding height of 52 cm, and three input files that include how the water fraction was varied for the bounding fissile material distribution.

This information is required to meet TS-R-1 671(a), 677, and 679.

4-4: Confirm earlier assumption that Content 1.5 was analyzed using the HET4 criticality model.

Table 4 for the response to RAI 5-7 states that the calculation results were based on the HET1 model. However, the supporting text in the response states that Content 1.5 was analyzed using the HET4 model. Please provide confirmation to the reviewer's assumptions that HET4 was used in the analysis.

This information is required to meet TS-R-1 671(a), 677, and 679.

4-5: Verify whether the arrays were analyzed in a dry condition or flooded, and include input files for ARRAY1 and ARRAY2.

Results provided in Section 8.7 of the criticality analysis state that the array of packages is assumed to be dry. However, Section 8.7.1 addresses the influence of water between the packages.

This information is required to meet TS-R-1 671(a), 677, and 679

References:

1. NUREG/CR-6802, "Recommendations for shielding evaluations for transport and storage packages," Oak Ridge National Laboratory, Oak Ridge, Tennessee, May, 2003.

2. L. B. Shappert (editor), “The Radioactive Materials Packaging Handbook,” ORNL/M-5003, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1998.
3. NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” Oak Ridge National Laboratory, Oak Ridge, Tennessee, March, 2003.