

## Followup RAIs for the NCS 45 Criticality Analysis

### **RAI1:**

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.

### **RAI2:**

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.

### **RAI3:**

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.

**RAI4:**

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.

**RAI5:**

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

**Additional RAI Comment: Review Summary:**

It should be noted that a number of conservatisms were utilized within the criticality safety analysis for this material. First, cladding and can materials were neglected in all single package and array cases. Secondly, in most cases the fissile material is assumed to be infinite in the axial direction and analyzed for a given maximum radius which is based on the physical size of the cavity or container. The H/U ratios were varied by varying the water fraction in the system. The results show that reactivity was the highest for water fractions resembling cases where the spacing was increased between rods to the maximum allowed diameters (e.g. 22 cm).

However, the reactivity in many of the analyzed cases is fairly high (**approaching 0.95**). Therefore, the reviewer has to develop an adequate understanding and degree of confidence in the methodology that was used to perform the calculations in the criticality analysis and adjoining calculation note.

**Followup RAIs for the NCS 45 Shielding Analysis****Requests for Additional Information Pertaining to the Shielding Safety Analysis of the NCS 45 Application**

RAI 7-1

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 Safety Analysis Report and update the shielding analysis (if necessary) based on updated source terms.

The applicant provides, in Tables 7-37 and 7-41 of the Safety Analysis, 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.

## RAI 7-2

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.

### RAI 7-3

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.

#### RAI 7-4

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.

#### RAI 7-5

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<sub>2</sub> 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 Safety Analysis Report, 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 Safety Analysis Report states that source terms were determined based on a fuel content of 4.5 w% U-235 in UO<sub>2</sub> compound with a 60,000 MWd/tU burnup and a 365 days of decay time. However, the Safety Analysis Report 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<sub>2</sub> 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.

#### RAI 7-6

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 Safety Analysis Report 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.

#### RAI 7-7

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 Safety Analysis Report 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.

#### RAI 7-8

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 Safety Analysis Report, the applicant describes the assumptions used in the shielding safety analysis under hypothetical accident conditions. It seems from the shielding analysis model provided Section 7.6 of the Safety Analysis Report 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 para. 727 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.

#### RAI 7-9

Provide justification for the use of different content definitions in different parts of the Safety Analysis Report 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.

#### RAI 7-10

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 Safety Analysis Report provides. 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.

#### RAI 7-11



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.

#### RAI 7-12

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.

Reference:

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.