Shielding SER for NCS-45 DOT Revalidation Review

X. SHIELDING EVALUATION

The purpose of the shielding review is to ensure that the package design meets the radiation limits set forth in the IAEA's regulations for radioactive materials transportation, TS-R-1. The review includes evaluation and verification of the applications shielding calculations contained in the section of the Safety Analysis Report (SAR) titled "Dose Rate Calculation" as well as descriptions of the design and technical drawings contained in the SAR.

Package Shielding Design

The D/4347/B(U)F-96 is designed for the transportation of irradiated uranium oxide in the form of fuel rods, fuel rod sections, pellets, as well as pellet scraps. The allowable contents are limited to Uranium oxide and Gadolinium in the form of Gd_2O_3, for contents 1.1, 1.2, 1.3, 1.4, and 1.5. Activated structural materials, cladding tubes, and surface contaminated parts from hot cells make up contents 5.1. Mixed loading of content 1 among each other as well as with content 5 is not admissible. Graphite or Beryllium is not allowed in the package. Table 1.1 of the Certificate of Approval describes the contents limits.

The package shielding design consists of concentric cylindrical stainless steel, lead, cement, and stainless steel shells in the radial direction. Stainless steel, lead, and tungsten are used to provide shielding at the top (lid) and the bottom of the cask. The packaging is open at both ends. Open ends of the body are closed by lid and bottom plugs using screwed flanges. Impact limiters are constructed of balsa and spruce wood in a stainless steel shell and fastened to the ends of the cask main body by 6 cylinder head screws. The design has four trunnions to accommodate handling of the cask. Two attached at each end of the cask.

The package consists of a payload cavity of maximum length of 462.5 cm and diameter of 22 cm, with either guide tubes to support fuel rods and/or a center axial cavity to axially align materials welded in cans. PWR or BWR fuel rods, scraps, and pellets with maximum burnup of 120 GWd/MTU and a minimum 120 day cooling time are admissible contents. The maximum payload is 100kg. The Contents with burnup less than 45 GWd/MTU are not required to be placed in welded cans unless other condition(s) requires doing so.

The design basis fuel is PWR or BWR fuel rods, scraps, and pellets with 60 GWd/MTU burnup and 100kg fuel payload. Fuel and Non-fuel materials are permitted. The applicant developed a set of empirical equations based on a similar package design (NCS R52). These equations include source strength, radiation type factor and content height factor. The users of these packages must determine the admissible quantity of the payload using these equations based on the source strength (per unit weight) and the TS-R-1 dose rate limits.

The staff reviewed the information presented in the SAR on the package design and found the methodology is not adequate because in the majority of the cases, as shown in Tables 7-5 and 7-6, the measured dose rate values are above the calculated values. In the worst case, the measured dose rate is 2.37 times of the calculated value. The staff requested the applicant to provide justification for the adequacy of these equations for determining the dose rates, the applicant states in its response that the user is required to determine the allowable contents based on the dose rate limits and the empirical equations described in the SAR and CoC.

In addition, the applicant presented its source term calculation results for the proposed contents in Tables 7-37 and 7-41 for gamma and neutron respectively. However, review of the data indicates that that these results are incorrect; they do not follow a well-understood relationship between source strengths and fuel burnup. The staff's independent calculations cannot confirm the data presented in Tables 7-37 and 7-41.

In its responses to the Requests for Additional Information regarding the source terms calculated by the applicant, the applicant provided data to show that the gamma source term versus burnup does exhibit linear relationship for cooling time greater than 5 years. The applicant's data however does not show the correct relationship between neutron source strength and fuel burnup. The applicant further states that "For low burnup rule of thumbs does not apply in full." However, the data presented in Table 7-41 do not support the same results as shown in the applicant's response to RAI 7-1 even for irradiated fuel with 10 years of cooling time.

Although it does not agree the approaches used in the package design, given the fact that the foreign competent authority has approved this package design, and the package design uses the TS-R-1 dose rate criteria as the design objective, and the packages are designed for exclusive use, the staff recommends the Department of Transportation to consider approval of this package design with significant reduction of the package content limit. Based on an analysis of the data provided in Table 7-5 (which are used in deriving the dose rate factors and are the bases for the NCS 45 package design), it seems that the average error in the measured versus calculated neutron and gamma dose rates is -21.5%. Therefore a twenty percents (20%) reduction of the design basis contents will provide a reasonable assurance that the package will meet the TS-R-1 dose rate limits. For irradiated fuel with higher burnup, the reduction should be proportional to the burnup with the factor derived from the ratio of the actual over design basis burnup (60 GWd/MTU), i.e.,

$$F = BU/60.$$

$$\text{load} = 0.8 * (\text{proposed content in Table 1.1})/F$$

With this payload content scaling down factor, fuel with 80 GWd/MTU burnup will be limited to

$$\text{load} = 100\text{ kg}/(80/60) * 0.8 = 60\text{ kg}.$$

Fuel with 90 GWd/MTU burnup will be limited to

$$\text{load} = 100\text{ kg}/(90/60) * 0.8 = 53.3\text{ kg}.$$

Other fuels follow. It is recommended that the Certificate of Approval includes this specific limit.

In addition, it is recommended that any fuel with burnup greater than 45 GWd/MTU should be sealed in welded cans and treated as damaged fuels because of the unknown property of the high burnup fuels.

The TS-R-1 regulations with shielding issues relevant to this cask are listed in the table below as well as the ability of this cask to meet these regulations.

Requirements for Shielding review of DOT revalidation

TS-R-1	TS-R-1 SUMMARY	Shielding Evaluation
501 (b) 502(a)	Shall be ensured that the effectiveness of its shielding and containment and, where necessary, the heat transfer characteristics and the effectiveness of the confinement system, are within the limits applicable to or specified for the approved design. For any package it shall be ensured that all the requirements specified in the relevant provisions of these Regulations have been satisfied.	The shielding properties of the cask materials were provided in Chapter 7 of the SAR and checked against independent sources, such as the Periodic Table of Elements, and the Radiological Health Handbook. The staff recommends significant reduction of the payload because the approach is inconsistent with the TS-R-1 required design certification approach.
526 526(a)	Specifies that the TI for a package, overpack, or freight container, or for unpackaged LSA-I or SCO-I, shall be the number derived in accordance with the procedure in paras. 526 - 527. Determine the maximum radiation level in units of mSv/h at a distance of 1 m from the external surfaces of the package, overpack, freight container, or unpackaged LSA-I and SCO-I. The value determined is multiplied by 100 and the resulting number is the transport index. For uranium and thorium ores and their concentrates, the maximum radiation level at any point 1 m from the external surface of the load may be takes as: .4 mSv/h for ores and physical concentrates of uranium and thorium; 0.3 mSv/h for chemical concentrates of thorium; 0.02 mSv/h for chemical concentrates of uranium, other than uranium hexafluoride.	This is an exclusive use package. At 2 meters from the external surface of the cask, the dose rate is not to exceed 0.1 mSv/h (10 mrem/h).
526(c)	The value obtained in (a) and (b) shall be rounded up to the first decimal place (e.g. 1.13 becomes I.2) except that a value of 0.05 or less may be considered as zero.	This is an exclusive use package.
530	The transport index of any package or overpack shall not exceed 10, nor shall the CSI of any package or overpack exceed 50 except for consignments under exclusive use.	This is an exclusive use package.
531 532	Sets the maximum radiation level on external surfaces of packages or overpacks except for those: transported under exclusive use by rail and road under the conditions specified in subpara. 572(a); or, under exclusive use and special arrangement by vessel or by air under the conditions specified in paras. 574 or 578. The maximum radiation level at any point on any external surface of a package or overpack shall not exceed 2 mSv/h. The maximum radiation level at any point on any external surface of a package under exclusive use shall not exceed 10 mSv/h.	Dose rate calculations and modeled data cannot be reproduced and verified. However, since this is a foreign competent authority approved design, it is for exclusive use, and the TS-R-1 dose rate limits are design objective, the staff recommends approval of this package with significant reduction of authorized payload.
601	Requirements for LSA-III material (601) LSA-III material shall be a solid of such a	Part 71 does not include LSA packages, and does not recognize fissile LSA materials.

	nature that if the entire contents of a package were subject to the test specified in para. 703 the activity in the water would not exceed 0.1 A2.	
645	A radiation shield which encloses a component of the package specified as a part of the containment system shall be so designed as to prevent the unintentional release of that component from the shield. Where the radiation shield and such component within it form a separate unit, the radiation shield shall be capable or being securely closed by a positive fastening device which is independent of any other packaging structure.	Not applicable because containment and shielding form a single unit.
646 646(b)	A package shall be so designed that if it were subjected to the tests specified in paras. 719-724 [Type A package tests], it would prevent: Loss of shielding integrity which would result in more than a 20% increase in the radiation level at any external surface of the package.	The HAC does rates are provided in Table 7-32 and 7-33.
651	A package shall be so designed that, under the ambient conditions specified in paras. 653 and 654, heat generated within the package by the radioactive contents shall not, under normal conditions of transport, as demonstrated by the tests in paras. 719-724, adversely affect the package in such a way that it would fail to meet the applicable requirements for containment and shielding if left unattended for period of one week. Particular attention shall be paid to the effects of heat, which may:	Temperatures calculated in chapter 5 at the package and inside the package are by far lower than the admissible temperature and melting temperatures of the packaging materials and of the radioactive contents (see chapter 5). The package consists of austenitic stainless steel with outstanding corrosion resistance. Humidity inside the package will not lead to corrosion. Ambient conditions are considered.
651(a) 651(b)	Alter the arrangement, the geometrical form or the physical state of the radioactive contents or, if the radioactive material is enclosed in a can or receptacle (for example, clad fuel elements), cause the can, receptacle or radioactive material to deform or melt; or Lessen the efficiency of the packaging through differential thermal expansion or cracking or melting of the radiation shielding material; or	All fuel with burnup above 62 GWd/MTU must be enclosed in a welded canister. The materials for these cans, and methods and drawings of construction are not given. Due to the unknown properties of the Zircaloy cladding above 45 GWd/MUT, any fuel rods with burnup higher than 45 GWd/MTU not 62 GWd/MUT should be placed in welded cans and treated as damaged fuel.
656(b)	The tests specified in paras. 726, 727 (b), 728 and 729 and the tests in paras: (i) 727I, when the package has a mass not greater than 500 kg, an overall density not greater than 1000 kg/m³ based on the external dimensions, and the radioactive contents greater than 1000 A_2 for not as special form radioactive material, or (ii) 727(a) for all other packages, it would: (i) retain sufficient shielding to ensure that the radiation level at 1 m from the surface of the package would not exceed 10 mSv/h with the maximum radioactive contents and (ii) restrict the accumulated loss of radioactive contents in a period of one week to not more than 20 A_2	Thermal insulation is protected by 25 mm thick stainless steel outer shell of the package. Drop tests show that the thermal insulation remains undamaged under accident conditions of transport (see chapter 4).

	for krypton-85 and not more than A_2 for all other radionuclides.	
807 (a)	a detailed description of the proposed radioactive contents with reference to their physical and chemical states and the nature of the radiation emitted;	Detailed descriptions of the allowable contents and limitations on the contents are included in the Certificate of Approval. However, these permissible contents are reduced by a the conditions specified in the CoC
807 (b)	a detailed statement of the design, including complete engineering drawings and schedules of materials and methods of manufacture;	Design details are provided in Chapter 3 and 9 of the SAR.
807 (c)	a statement of the tests which have been done and their results, or evidence based on calculative methods or other evidence that the design is adequate to meet the applicable requirements;	Tests performed and results of the tests are described in Chapter 4 of the SAR.
807 (d)	the proposed operating and maintenance instructions for the use of the packaging;	The procedure for drying the cavity of the cask and fuel after loading was checked and found to be acceptable.
807 (f)	where the proposed radioactive contents are irradiated fuel, the applicant shall state and justify any assumption in the safety analysis relating to the characteristics of the fuel and describe any preshipment measurement required by para. 674(b);	Information about the irradiated fuels including the irradiation history are not available per the applicant's response to staff's Request for Additional Information states: "The information requested in RAI 7-3 cannot be provided because of the variety of the possible contents."