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PRELIMINARY SHIELDING AND CRITICALITY SAFETY ANALYSES OF A DUAL PURPOSE CASK FOR SPENT FUEL FROM LATIN AMERICAN RESEARCH REACTORS

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ABSTRACT

Shielding and criticality safety calculations carried out for an interim storage and transportation cask are presented. Such dual purpose cask is being designed for spent fuel elements of research reactors in the Latin American region. The Monte Carlo transport code MCNP4B was utilized for the criticality safety analysis part and SCALE for shielding. The analyses considered only two types of fuel assemblies utilized in the region and the results show that both types can be loaded in the designed cask and baskets in compliance with the safety criteria.

1. INTRODUCTION

The IAEA Technical Cooperation Project RLA/4/018 “Management of Spent Fuel from Research Reactors” started in 2001. It constitutes a joint effort of Latin American nuclear institutions from Argentina, Brazil, Chile, Mexico, and Peru to accomplish the following objectives: “to define the basic conditions for a regional strategy for managing spent fuel which will provide solutions that are in the economic and technological realities of the countries involved, and in particular, to determine what is needed for the temporary wet and dry storage of spent fuel from the research reactors in the countries of the Latin American region”.

Such work gets considerable importance since the USA spent fuel take-back program will be ended in May, 2006. After that the Latin American research reactors operators will need to identify and assess all possible options to deal with their spent fuel by themselves.

Part of the project consists on to design a cask for interim storage and transportation of the spent fuel produced by these research reactors. In Latin America there are two main types of research reactors under operation: TRIGA and MTR reactors. Such concern is being taken into account in the cask design that must be suitable for the different fuel elements used in these reactors.

The shielding analysis of the cask is being carried out with SCALE[1] package, versions 4.4A and 5. SCALE is a codes system for licensing purposes well known and frequently used for criticality safety, source terms determination, shielding and Heatload analyses of similar casks manufactured by different vendors around the world. The criticality safety analysis of the cask was carried out with the Monte Carlo transport code MCNP4B[2].

2. DESIGN CRITERIA

As recommended by International Atomic Energy Agency (IAEA) [3, 4] the shielding and criticality safety criteria to the cask can be summarized as follows:

The maximum dose values at the highest intensity point must be, for the sum of the gamma and neutron dose rates, lower than

> 10000 μSv/h (1 Rem/h) - local dose rate on the surface of the cask which can be easily touched during transport under exclusive use.

Concerning criticality safety the calculated neutron multiplication factor using Monte Carlo transport methods must be for normal conditions as well as under accident conditions lower than 0.95. Such condition can be mathematically expressed as,

$$(k_{eff} + 2\sigma + \beta) < 0.95,$$

where k_{eff} is the effective neutron multiplication factor, σ is the standard deviation of the k_{eff} and β is the bias. This condition must be guaranteed under an optimally moderated situation that will be assumed here as corresponding to the cask totally water flooded. The bias is assumed zero in this text since its determination depends of benchmarks which were not evaluated at the moment. The estimation of the bias shall be the subject of another work.

3. FUEL ELEMENTS DATA

The reference [5] lists 16 research critical facilities currently under operation in the 5 Latin American countries participants of the IAEA-RLA/4/018 project. More than a dozen and a half of different fuel element types are (or were) utilized by these research reactors. From all these fuel elements only two types were chosen to be analyzed in this text. The first one is the low enrichment TRIGA type, indeed the fuel elements used in the Brazilian TRIGA IPR-R1 reactor [6, 7]. The another fuel element type considered refers to MTR fuel elements, box type, actually the standard U3O8-Al type manufactured by CNEA (Argentinean Atomic Energy Commission) for its RA-3 research reactor [8-10]. The criterion to choose these two types of fuel elements among so many was due to the higher abundance of fuel elements similar to them used in the region (the majority of TRIGA fuel utilized by Brazil and Mexico are low enrichment TRIGA and concerning MTR fuel Argentina, Brazil and Peru utilize the U3O8-Al fuel). Therefore, an analysis of these two types of fuel elements gives information useful for the wider number of countries. Needless to say that for licensing purposes in the different countries the specific characteristics of the other fuel elements must be taken into account.

The main characteristics of the two types of fuel elements are summarized in tables I and II and in figures 1 and 2. The RA-3 standard fuel elements have 19 fuel plates fueled with U_3O_8-Al in Aluminum cladding assembled in an Aluminum structure and the IPR-R1 (TRIGA) fuel elements are cylindrical bars fueled with a metallic Uranium and Zirconium-Hydride homogenous alloy also in Aluminum cladding.

Table I. Compositions of the Fuels.

	U_3O_8-Al MTR Fuel Elements (RA-3)	TRIGA Fuel Elements (IPR-R1)
Mass of U3O8 (g)	1743.2	----
Mass of HZr (g)	----	2250
Mass of H (g)	----	24.5

Mass of Uranium (g)	1475.7	188
Mass of U235 (g)	290.7	37.2
Mass of Aluminum (g)	626.1	----
Enrichment (% mass)	19.75	19.81
Density (g/cm3)	4.8	6,2

Table II. Dimensions and Materials Data

Component	Dimension (cm)	Material
MTR Fuel Plates		
Active Zone		
Thickness	0.07	U308-Al (ρ=4.8 g/cm3)
Width	6.0	U308-Al
Height	61.5	U308-Al
Cladding thickness	0.04	Aluminum (ρ=2.7 g/cm3)
Plates Height	65.5	Aluminum
Others	as figure 1	
TRIGA Fuel Elements		
Active Zone		
Outer diameter	3.73	
Fuel Height	35.56	
Fuel diameter	3.56	U-ZrH (ρ=6.2 g/cm3)
Cladding thickness	0.076	Aluminum (ρ=2.7 g/cm3)
Length	72.24	
Axial reflectors	10.16	Graphite (ρ=1.7 g/cm3)
Others	as figure 2	

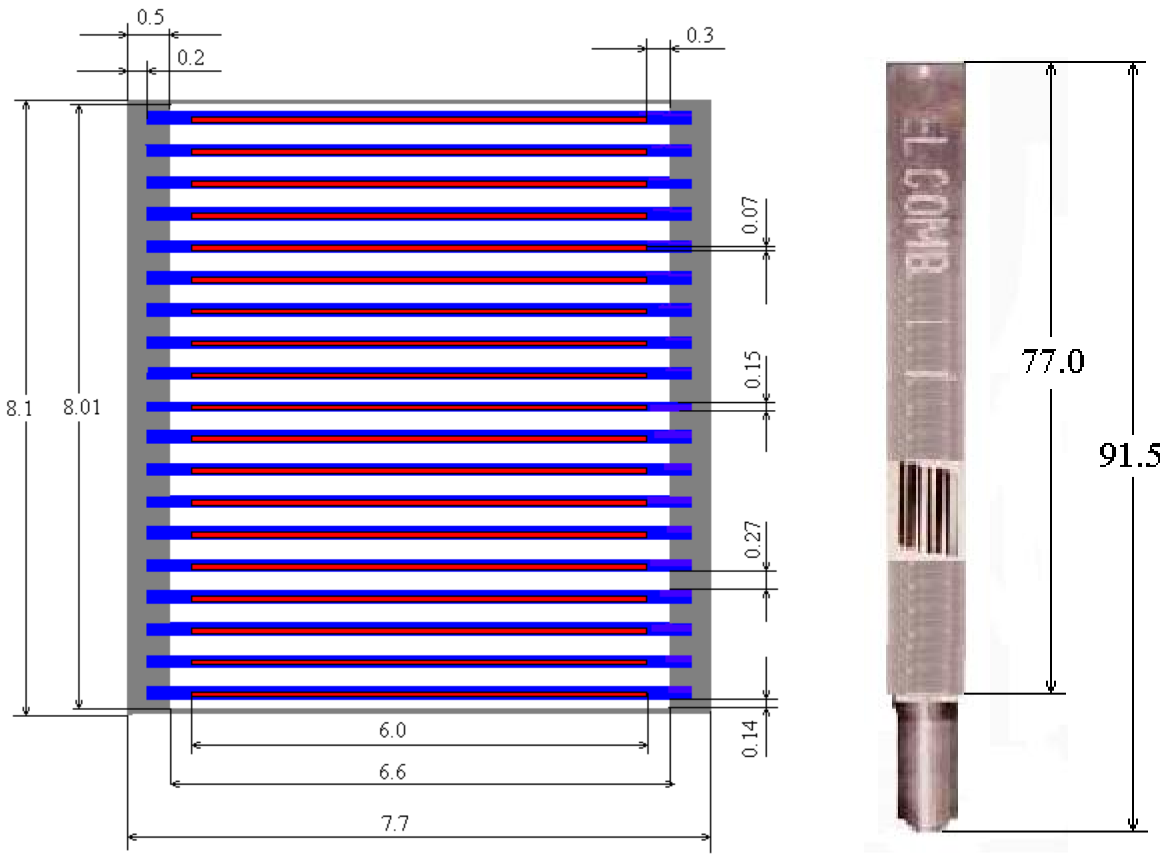


Figure1. U_3O_8-Al MTR fuel element

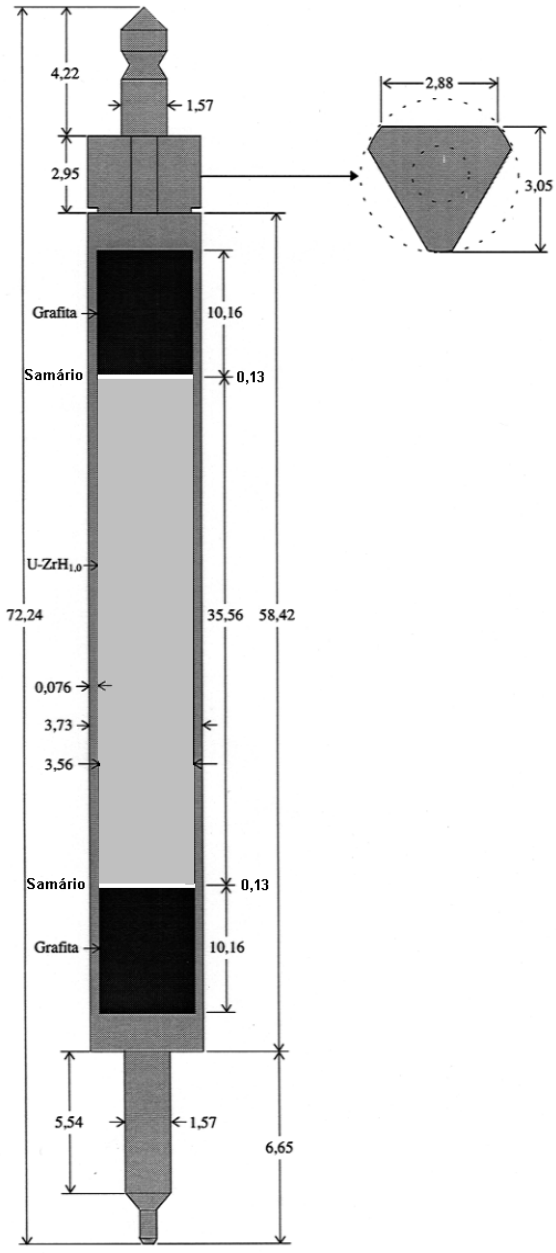


Figure 2. TRIGA fuel element

4. CASK GEOMETRY AND MATERIALS MODELING

The geometry and materials modeling of the cask follows the characteristics showed in the figure 3. All dimensions in the figure are in millimeters and no component tolerances are taken into account. It means that these dimensions shall be the minimum accepted for manufacturing processes. The cask has an inner and an outer shell (liner) made with stainless steel ANSI 304. A lead shield in between the shells completes the model.

Due to the different types of fuel elements used in the region many baskets shall be designed. Currently, the calculations considered only two types of basket, both made with SS304. One for MTR elements, which hosts 21 fuel elements (figure 3) and another basket for TRIGA with 78 locations (see figure 4).

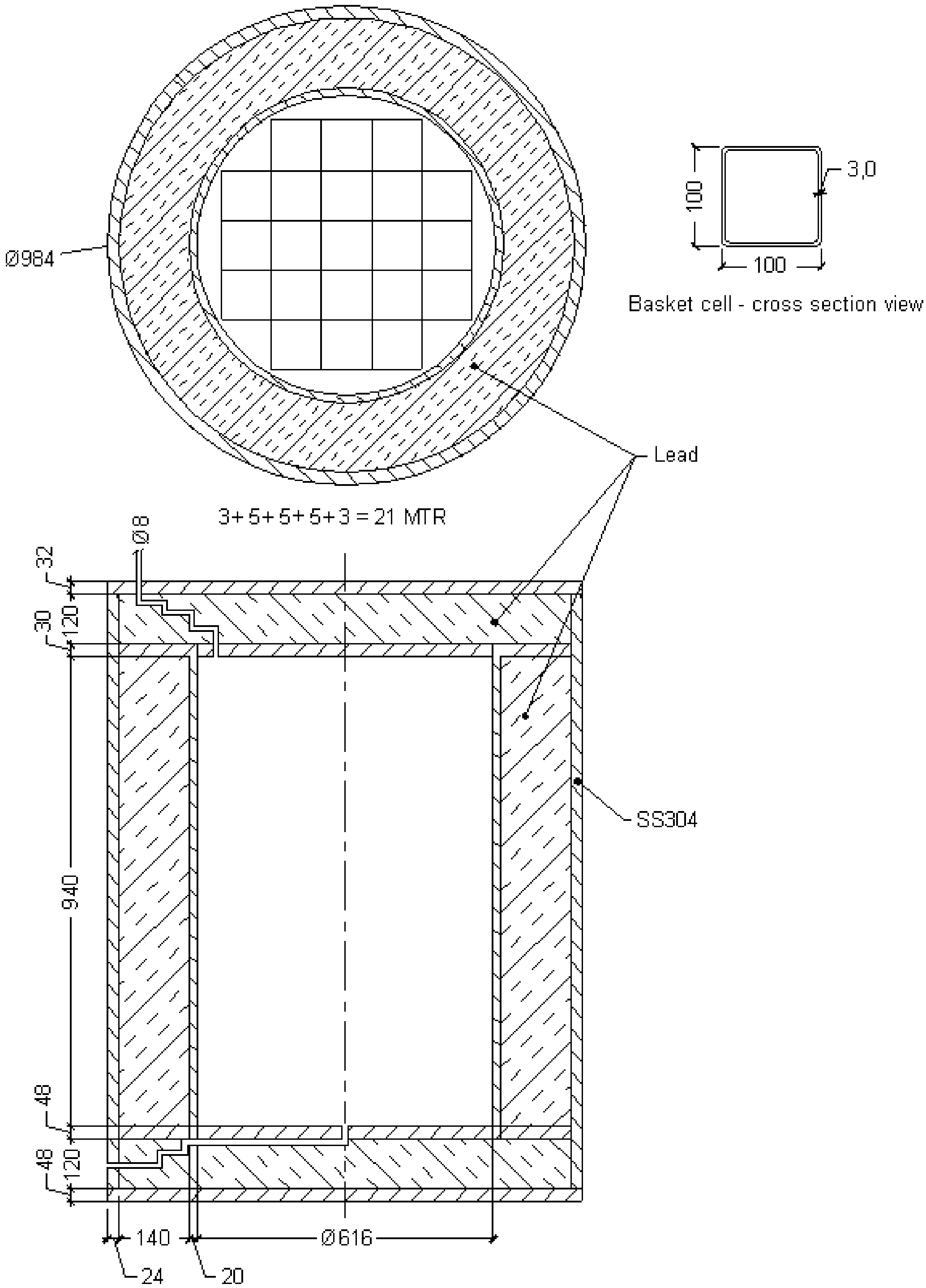
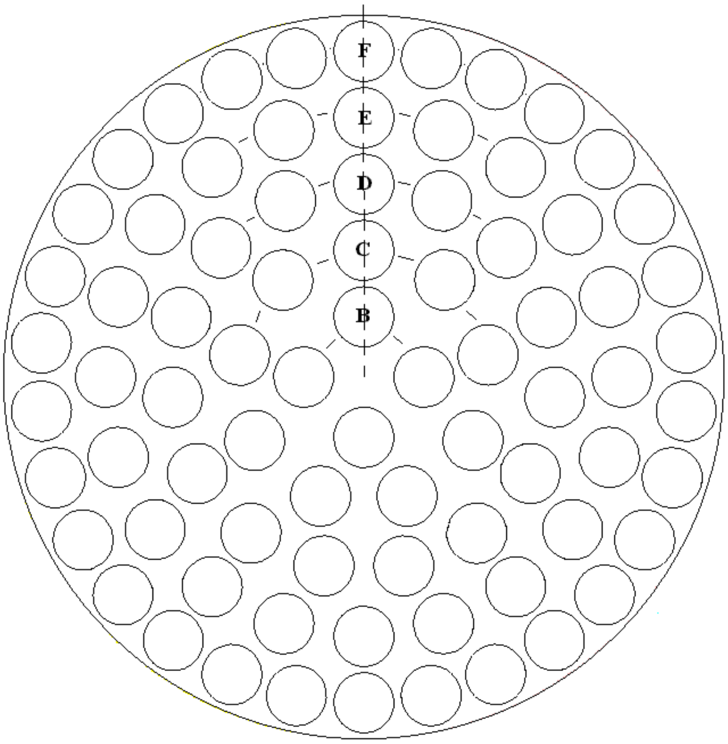


Figure 3. Schematic drawing of the cask with the 21-MTR basket



Radius (cm)

RB = 5.00 RC = 10.50
RD = 16.00 RE = 21.50
RF = 27.00 Pin Radius = 2.50

Number of Locations

B = 4 C = 9
D = 15 E = 20
F = 30

Figure 4. Schematic drawing of the TRIGA-basket

5. RADIATION SOURCE TERMS DETERMINATION

Two types of radiation sources were determined. The first one refers to the U_3O_8-Al MTR fuel elements. The another is for TRIGA type. The ORIGEN-ARP module of SCALE package was used to determine the gamma and neutron sources using libraries produced using the SAS2 module.

The gamma and neutron sources were estimated considering a burnup of 50% (U235 depleted) for MTR and 25% for TRIGA fuel. Furthermore, a 5 (five) years decay time (cooling time) was assumed in the model.

The gamma and neutron spectra as well as the source strengths can be seen in table III. Those values are per fuel element, it means that in order to get the total source strength inside the cask the values in the table III must be multiplied by the number of fuel elements. The energy group structure for neutrons and gamma are as the 27N-18Couple SCALE library.

Gamma with low energy, which are strongly shielded, are the majority and the high energy gamma are in minor amount. It is a favorable condition from shielding point of view. The same is observed for neutrons. The gamma and neutron source intensity is assumed flattened over the entire fuel element active length (that is a short length). Due to the weak neutron source intensity after five years cooling time no neutron-induced gamma radiation is assumed. Neutron source considers the contribution of spontaneous fission neutrons and neutrons from the α-n reactions in the oxygen of the fuel.

Table III. Calculated neutron and gamma source for MTR and TRIGA fuel

Energy (MeV)	Gamma (photons/s/element)		Energy (MeV)	Neutrons (neutrons/s/element)	
	MTR	TRIGA		MTR	TRIGA
10.0 to 8.0	2.8249E+00	1.0	20.0 to 6.43	8.018E+01	1.342E+00
8.0 to 6.5	1.3357E+01	1.0	6.43 to 3.0	1.263E+03	2.563E+01
6.5 to 5.0	6.8483E+01	1.3068E+00	3.0 to 1.85	2.004E+03	4.627E+01
5.0 to 4.0	1.7173E+02	3.2915E+00	1.85 to 1.40	8.357E+02	1.735E+01
4.0 to 3.0	2.9648E+07	6.1579E+05	1.40 to 0.90	9.196E+02	1.725E+01
3.0 to 2.5	2.6175E+08	5.4300E+06	0.90 to 0.40	8.828E+02	1.530E+01
2.5 to 2.0	3.3792E+10	6.9080E+08	0.40 to 0.10	1.711E+02	2.949E+00
2.0 to 1.66	5.6799E+09	1.5103E+08			
1.66 to 1.33	7.3329E+10	1.6087E+09			
1.33 to 1.0	1.8011E+11	5.0044E+09			
1.0 to 0.8	7.4752E+11	1.4928E+10			
0.8 to 0.6	1.2597E+13	6.8141E+11			
0.6 to 0.4	2.0304E+12	4.7645E+10			
0.4 to 0.3	5.4545E+11	2.4711E+10			
0.3 to 0.2	7.3889E+11	3.3934E+10			
0.2 to 0.1	2.6894E+12	1.1497E+11			
0.1 to 0.05	3.3567E+12	1.5790E+11			
0.05 to 0.01	1.1223E+13	5.3731E+11			
Total	3.4221E+13	1.6203E+12		6.157E+03	1.261E+02

6. CALCULATION METHOD

6.1. Shielding

The shielding calculations were performed using the SAS4 module of SCALE4.4A and of SCALE5. The Shielding Analysis Sequence No. 4 (SAS4) performs a three-dimensional Monte Carlo shielding analysis of a nuclear fuel transport or storage cask using an automated biasing procedure. In a Monte Carlo shielding analysis of a deep-penetration problem such as a spent fuel cask, variance reduction techniques must be used to calculate reasonably good results at an affordable cost. Generation of biasing parameters and application of the parameters to solve a particular problem are no trivial tasks. The entire procedure for cross-section preparation, adjoint flux calculation, automatic generation of Monte Carlo biasing parameters, and a Monte Carlo calculation has been implemented in this control module to provide calculated radiation dose levels exterior to the cask at a reasonable computational cost. The response function (dose factor) utilized in the calculation is the Neutron and Gamma Ray Flux-to-Dose Rate Factors ANSI/ANS-6.1.1, 1977.

As showed in figure 3 there are two holes (8 mm diameter) crossing the cask walls. One is located in the lid and is designed for cask pressurization with inert gas. The another hole is for the drainage of water and goes through the cask bottom and lower side wall. These holes may lead to the local increasing of dose rates around them. An usual solution to avoid such hot points is to design not straight penetrations by warping of the tubes and consequently breaking the free path of the radiation. Nevertheless, this approach brings huge complications to the calculations of shielding.

The methodology used for automated bias generation in SAS4 is very effective for cylindrical cask systems; however, it has a number of limitations arising from the use of one-dimensional (1-D) adjoint fluxes in the creation of automated biases. One such limitation is exactly the estimation of particles streaming through voids. Furthermore, it has limitations on geometry modeling. Therefore, the shielding analysis strategy will follow the approach of a minimum shielding thickness for the cask penetrations. It means that a minimum Lead thickness (enough thick to reduce the dose rates below the shielding criterion) must always exist between a void straight path and the outermost surface of the cask (see figure 5). Such approach is strongly conservative.

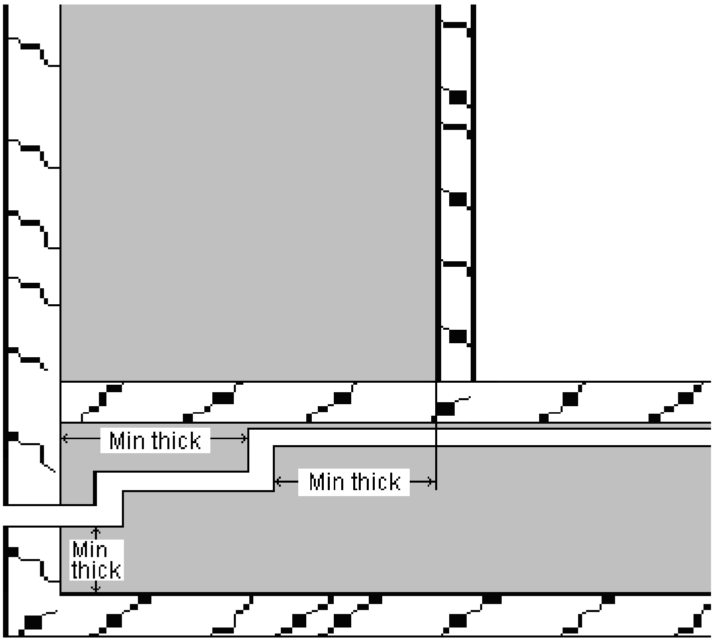


Figure 5. Scheme of the Minimum Lead Thickness for the Cask penetrations

Regarding the fuel region simulation the homogeneous SAS4 model, IGO = 0, was utilized. It means a homogeneous cylindrical source zone. The cask cavity has 30.8 cm radius and 94 cm height. However, the active zones of fuel elements have 61.5 cm height and 38.1 cm height, respectively, for MTR and TRIGA fuel and they are located at the cask axial mid-plane. Therefore, the model assumes two equally spaced void axial zones in the bottom and top of the cask cavity due to the negligible activation of the Aluminum end fittings of the fuel elements. It is important to note that the active fuel zone must be kept as symmetrically close of the cask axial mid-plane as possible since the "square of the distance factor" is not negligible and may lead to dose rates increasing near the top or the bottom of the cask if the active zone is closer to one or the another. No radial void is assumed. Finally, the source material fulfilling the source zone is modeled with one third of the fuel density and the concentration of heavy metal (Uranium) in the fuel is reduced by the same factor in order to reduce conservatively the self-shielding in source zone.

6.2. Criticality Safety

The main concern of the cask criticality safety analysis is to guarantee that the nuclear fuel inside the cask will be maintained safely sub-critical under normal conditions of operation and hypothetical accident conditions. The cask leak-tight containment shall be able to prevent water infiltration into the cask cavity. This is applicable for normal conditions and also after the cask suffers the type B tests.

The criticality safety calculations were performed using the MCNP4B code. It is a multipurpose Monte Carlo program used for stochastic simulation of particles transport (neutrons, photons and electrons). MCNP is a 3D code and has graphical resources very useful for modeling visualization.

In the model every criticality calculation assumes that the fuel elements are fresh, which means that no burnup is considered. This is a very conservative hypothesis as such situation definitively is not the standard for a spent fuel elements cask. The masses of fissile material in the fresh fuel elements were informed by the reactors operators. Also conservatively no poisoning effect or material is considered and in particular for TRIGA fuel the Samarium burnable poison discs are not taken into account.

All calculations were carried out using the point-wise cross-sections library based on ENDF/B-VI data. The calculations were performed considering room temperature, equal to 293 Kelvin. No temperature corrections were applied to the MCNP cross-section data. A very accurate geometrical model for each fuel element in the basket is used. All simulations used 5000 neutron histories and 750 cycles, skipping the 50 first cycles in order to avoid fluctuations coming from misdistribution of neutron source points.

Regarding the neutrons moderation four situations were analyzed. The first is the normal condition of operation in which the cask is fully dry and just air surrounds the fuel elements. The three others consist on accident conditions in which the cask is totally water flooded, being one corresponding to only the cavity flooded, a second in which the cavity is flooded and also the whole cask is surrounded by a 30 cm thick layer of water and finally, only for MTR21 basket the third accident situation was simulated in which the dimensions of the internal basket cells were reduced from 10.0 cm to 8.7 cm in order to simulate the hypothetical situation in which the fuel elements are as close to each other as possible.

7. RESULTS

7.1. Shielding

The shielding calculation results are summarized from table IV to VI. Tables IV and V present the dose values at the highest intensity point at cask outermost surfaces at side wall and lid (the top is worst situation than bottom as it has thinner stainless steel liners) for both fuel element types evaluated. Results shown in table IV were obtained using SAS4 and refer to TRIGA basket while table V refers to MTR21 basket and were obtained neglecting the two cask penetrations. The neutron dose contribution to the total dose rates is not significant. The main contribution comes from gamma radiation, as expected, since the analysis was carried out considering a five years cooling time. The highest dose rates occur at the top surface (lid) of the cask. In any case such values are well below the limit for shielding criterion.

Table VI shows dose rates for different Lead thicknesses and the cask loaded with basket MTR21 (TRIGA fuel was neglected as it has weaker radiation strength). The main goal of such analysis was to find the minimum Lead thickness to guarantee dose rates values below the shielding criterion around the two cask penetrations (see figures 3 and 5). These minimum Lead thickness must be at least 8.0 centimeters radial and 7.0 centimeters axially between each void straight path and the cask outermost surface.

Table IV. SAS4 dose rates for basket with 78 TRIGA fuel elements

		Dose Rates at Side Surface (µSv/h)	
Case	Source	SCALE4.4A	SCALE5
N_rad_78	Neutron	< 1	< 1
G_rad782004	Gamma	19	18
	Total	20	19
		Dose Rates at Lid Surface (µSv/h)	
Case	Source	SCALE4.4A	SCALE5
N_bot_78	Neutron	< 1	< 1
G_bot782004	Gamma	32	26
	Total	33	27

Table V. SAS4 dose rates for basket with 21 MTR fuel elements

		Dose Rates at Side Surface (µSv/h)	
Case	Source	SCALE4.4A	SCALE5
N_rad200408	Neutron	6	6
G_rad200414	Gamma	186	173
	Total	192	179
		Dose Rates at Lid Surface (µSv/h)	
Case	Source	SCALE4.4A	SCALE5
N_bot200412	Neutron	4	3

G_bot2004	Gamma	367	280
	Total	371	283

Table VI. SAS4 dose rates for different Lead thicknesses (21 MTR fuel elements)

	Side Surface - Radial					
Lead thickness (cm)	14.0		8.0		7.0	
SCALE version	4.4A	5	4.4A	5	4.4A	5
Gamma Dose Rates (μSv/h)	186	173	5686	5522	11146	10947

	Lid Surface - Axial									
Lead thickness (cm)	12.0		10.0		8.0		7.0		6.0	
SCALE version	4.4A	5	4.4A	5	4.4A	5	4.4A	5	4.4A	5
Gamma Dose Rates (μSv/h)	367	280	1230	836	3329	2761	6804	5081	11766	10060

7.2. Criticality Safety

The criticality calculations results are summarized in tables VII and VIII. They present the $k_{eff} \pm 2\sigma$ to the cask loaded with the TRIGA basket and with the MTR basket. In both cases the results refer to moderation conditions mentioned in section 6.2.

The highest k_{eff} is to the cask loaded with MTR21 basket, inside and outside flooded and with the reduced dimensions of the internal basket cells. One can note that under normal conditions, in which the cask is dry, the k_{eff} of the MTR basket is too small while for TRIGA basket it is almost a half of the value to the cask flooded. Such fact represents well the importance of the Zirconium-hydride alloy as moderator in the TRIGA fuel.

Table VII. Criticality results for TRIGA Fuel

Case	($k_{eff} \pm 2\sigma$)	Remarks
CA78AR	0.41213 ± 0.00066	78 TRIGA elements and Air inside the cask
CA78AG	0.87049 ± 0.00072	78 TRIGA elements and water inside the cask

CA78AGF	0.87165 ± 0.00070	78 TRIGA elements and water inside and outside the cask

Table VIII. Criticality results for MTR Fuel

Case	($k_{eff} \pm 2\sigma$)	Remarks
CA21AR	0.05073 ± 0.00008	21 MTR elements and air inside the cask
CA21AG	0.86348 ± 0.00084	21 MTR elements and water inside the cask
CA21AGF	0.86412 ± 0.00084	21 MTR elements and water inside and outside the cask
CA21AX	0.89890 ± 0.00082	21 MTR elements, water inside and outside the cask and 8.7 cm side length of the internal basket cells

In any case, concerning the criticality safety criterion, it is guaranteed that for both fuel element types and also for both baskets here analyzed the cask can be loaded and unloaded safely. Thus, the fuel elements may also be stored or transported safely sub-critical even in the hypothetical situation of the cask to be water flooded.

As a design requirement the cask and the basket shall keep their integrity after the type B tests. Nevertheless, there may be local deformations of the cask inner liner and basket. This situation must also be analyzed, concerning both criticality safety and shielding, but it makes sense only after the type B tests that are being planned for a half scale model of this dual purpose cask. Moreover, if in the future the nuclear policies of the participant countries lead to the construction of a cask prototype a safety analysis report will shall be written and then, for such purpose, further calculations will be necessary in order to analyze the other research reactors fuel element types currently used in Latin America. Finally, also and only for the purpose of a real safety analysis report, benchmarks that lead to determination of bias and errors related to the calculation methodologies are necessary.

8. CONCLUSIONS

The SCALE package was used for radiation source terms determination and shielding calculations of a Latin American dual purpose cask and MCNP4B for the criticality safety analysis. The analyses just considered two types of fuel elements (one TRIGA type and another MTR type) among several utilized in the Latin American research reactors. Moreover, two types of baskets were considered for loading of these fuel elements in the cask, one for 21 MTR elements and another for 78 TRIGA elements. In any case the results show compliance with the shielding and criticality safety criteria.

The simulations show that the highest dose rates occur at the lid (top surface) of the cask. The dose rates for TRIGA fuel are well below the values for MTR fuel. In the worst case the total dose rate reaches just a few hundreds of μSv/h, therefore, well below the limit for shielding criterion. In order to bypass limitations in the methodology for shielding calculations around the two cask holes a minimum Lead thickness approach was utilized and it was found that at least a Lead layer of 8.0 centimeters radial and 7.0 centimeters axially must be kept between each void straight path and the cask outermost surface.

Concerning the criticality safety analysis the sub-criticality is guaranteed for the cask loaded with both baskets and both fuel element types. The simulations of accident conditions in which the cask is totally water flooded and also outside surrounded by a 30 cm thick layer of water showed compliance with the criticality safety criterion. Actually, such situation means that even the underwater loading and unloading of fuel elements can be an operational alternative to be considered. Additionally, an extreme accident hypothesis was considered in which the flooded cask is modeled loaded with the MTR21 basket cells having the dimensions reduced from 10.0 cm to 8.7 cm and the k_{eff} still remains below 0.9. Thus, these two types of fuel elements can be safely stored or transported by the cask under the point of view of shielding and criticality safety.

ACKNOWLEDGEMENTS

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