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Review

# Shielding calculation and criticality safety analysis of spent fuel transportation cask in research reactors

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## HIGHLIGHTS

- Shielding calculation was carried out for general material testing reactor (MTR) research reactors interim storage.
- Criticality safety analysis was carried out for general MTR research reactors relevant transportation cask.
- The MCNP5 code was used for shielding calculation and criticality safety analysis.
- The ORIGEN2.1 code was used for source term calculations.

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## ABSTRACT

In this study, shielding calculation and criticality safety analysis were carried out for general material testing reactor (MTR) research reactors interim storage and relevant transportation cask. During these processes, three major terms were considered: source term, shielding, and criticality calculations. The Monte Carlo transport code MCNP5 was used for shielding calculation and criticality safety analysis and ORIGEN2.1 code for source term calculation. According to the results obtained, a cylindrical cask with body, top, and bottom thicknesses of 18, 13, and 13 cm, respectively, was accepted as the dual-purpose cask. Furthermore, it is shown that the total dose rates are below the normal transport criteria that meet the standards specified.



**Contents**



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## 1. Introduction

Radioactive materials are used in nuclear reactors to produce either electricity or radioactive waste. These radioactive wastes should be carefully stored and disposed of after conditioning and treatment. Thus, one of the most important challenges in nuclear industry is the management of the spent fuel and radioactive wastes. This study is focused on the spent fuel of research reactors (IAEA, 2006, 2000).

Spent fuels transported in the cask have exceptionally high radioactivity. Thus, all transportation casks must be designed to regulate radiation protection (Safety Series No. 6, 1985; Standard Safety Series, 2003, 2011).

The ORIGEN2.1 code (Croff, 1980) is used for source term calculation of typical material testing reactor (MTR) fuel used with a burn-up of 60%, expecting fission products and higher-order actinides during irradiation in a research reactor in accordance with Tehran Research Reactor (TRR) data (TRR, 2007). Data related to radioactive nuclides were obtained from ENDF/B-IV libraries, which were used for the depletion calculations. Furthermore, ORIGEN2.1 generates an 18-group gamma-ray source spectrum for fission and activation products of light-structure materials and actinides of heavy metals. Moreover, a 5-year decay cooling time was adopted in the model (Croff, 1980).

The MCNP5 code (Oak Ridge National Laboratory, 2003) was used to calculate neutron effective multiplication coefficient ($k_{eff}$) and collect information on gamma dose rate parameters for MTR fuel in TRR.

Therefore, the purpose of this study is to design transportation casks for optimum shielding and minimal outside dose rate below the levels recommended by international standards.

Analysis was carried out, and simultaneously sub criticality of the cask and its content was ensured at all operating conditions.

In order to obtain the desired results, TRR fuel information was used as a typical MTR fuel for a case study.

## 2. Material and methods

### 2.1. Description of MTR Fuel Elements

The TRR is a 5-MW reactor with water acting as the moderator, coolant, and shield, and a heterogeneous solid as the fuel. The reactor core is composed of MTR-type spent fuel assemblies. Initially, high-enriched uranium (HEU) was used as the fuel, which was replaced by low-enriched uranium (LEU) during refueling. In this study, low-enriched spent fuel with a 5-year cooling time is used (TRR, 2007).

As a general MTR fuel element, 20% enriched LEU fuel elements of TRR is considered. Detailed information on this type of fuel, as well as information on other aspects of the TRR is provided elsewhere (TRR, 2007).

Characteristics of fuel composition and aluminum cladding are presented in Table 1, and LEU–SFE (standard fuel elements) fuel geometry is shown in Fig. 1.

### 2.2. Design criteria

Shielding and criticality safety criteria of the spent fuel cask recommended by the International Atomic Energy Agency (IAEA) are as follows (Kang et al., 1988; Safety Series N. 37, 1985; Hiland and Taylor, 2005):

Nuclear criticality safety criterion for neutron effective multiplication coefficient ($k_{eff}$), which was calculated using Monte Carlo transport methods, must be $< 0.95$ in most conditions and 0.98 in extreme conditions. This condition must be guaranteed under an

**Table 1**
Standard fuel elements and SFE cladding data.

| | | |
|---|---|---|
| Fuel elements | $^{235}U$ per SFE | 2990 g |
| | $^{235}U$ per fuel plate | 76 g |
| The composition of the fuel elements | Enriched $U_3O_8$ | 20% enriched uranium |
| | Uranium density | 2.9617 g/cm$^3$ |
| | Weight percent | $^{235}U$, 12.45%<br>$^{238}U$, 49.78%<br>O, 11.18%<br>Al, 26.59% |
| Cladding | Al | 99.6%<br>2.7 g/cm$^3$ |

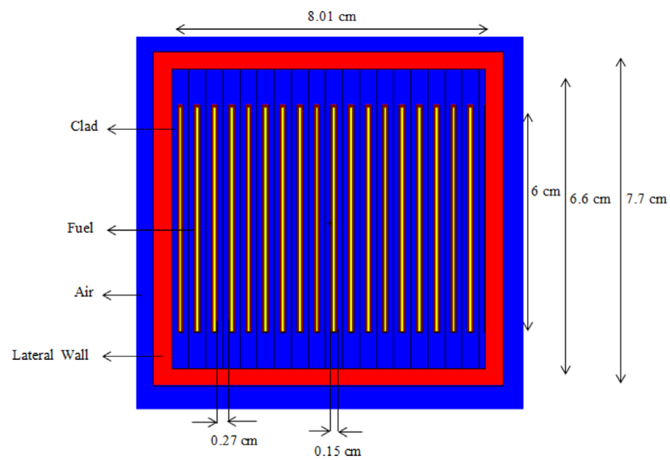


**Fig. 1.** Fuel element assembly.

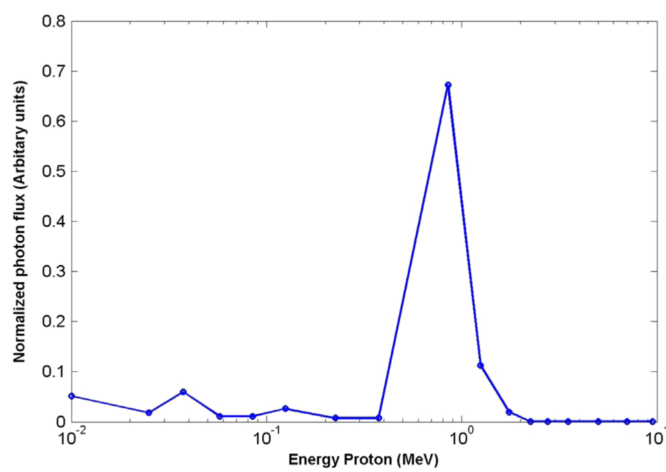


**Fig. 2.** Normalized photon flux spectrum after 5-year cooling time.

**Table 2**
Number of neutrons produced per second per spent fuel element after 5-year cooling time due to ($\alpha$, $n$) reaction.

| Radionuclides | Per fuel element (neutron/s) |
|---|---|
| Pu-238 | 3.85E+03 |
| Pu-239 | 3.05E+02 |
| Pu-240 | 4.12E+02 |
| Am-241 | 6.84E+02 |
| Cm-242 | 1.08E+01 |
| Cm-244 | 1.75E+02 |
| Total | 5.45E+03 |