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Radiation and criticality safety analyses for the highly-enriched uranium core removal from a research reactor

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HIGHLIGHTS

- A transfer shield was designed to remove the highly-enriched uranium core from a research reactor.
- Calculations were performed to determine the core's fission product inventory and radioactivity.
- Transfer shield reduced dose rates from 29 Sv/h to 8 mSv/h.
- Maximum individual dose was 0.84 mSv.
- Criticality analysis showed that the core was sufficiently subcritical in all situations.

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ABSTRACT

Analysis was performed to estimate radiation levels during removal and packaging of the highly-enriched uranium core of the JM-1 SLOWPOKE-2 research reactor. Due to severe limitations of space in and around the reactor pool, the core could not be removed in the conventional manner as was done for previous SLOWPOKE defuelling operations. A transfer shield, with a balance between shielding efficacy, volume and weight was designed. Fuel depletion, Monte Carlo shielding and criticality calculations were performed. Comparisons of measured and calculated dose rates as well as results of the criticality safety assessment are presented. The designed transfer shield reduced the calculated unshielded dose rate from 29 Sv/h to 8 mSv/h. The maximum calculated effective neutron multiplication factor of approximately 0.89 was below the 0.91 upper subcritical limit.

1. Introduction

The Jamaican SLOWPOKE-2 Research Reactor (JM-1) is a 20 kW tank-in-pool type reactor designed by the Atomic Energy of Canada Ltd (AECL), owned and operated by the University of the West Indies, Mona Campus, at the International Centre for Environmental and Nuclear Sciences (ICENS). JM-1 was commissioned in 1984 with 0.888 kg of uranium-aluminum alloy fuel enriched to approximately 93% ^{235}U. In 2009, plans were made to convert the reactor from highly-enriched uranium (HEU) to low-enriched uranium (LEU) through the then Global Threat Reduction Initiative (GTRI), now the Material Management and Minimization program with assistance from the International Atomic Energy Agency (IAEA).

Due to space limitations in and around the reactor pool and the inability to remove the core in the conventional manner, an assembly had to be designed to transfer the irradiated HEU core from the reactor container to the intermediate transfer system (ITS) of the transportation cask used to ship the core. The core removal procedure had to provide a balance between shielding efficacy, size and weight of the transfer shield to minimize radiological exposure of personnel, while making the transfer shield easy to manipulate. The JM-1 fuel cage is approximately 220 mm in diameter and in height and consists of upper and lower flanges connected by a hollow central spindle with bayonet slots that provided the locking mechanism for the core removal tool. The HEU core consisted of 296 fuel pins, each 5.23 mm in diameter and had been irradiated for 31 years prior to removal. This paper presents the analysis performed to determine the required amount of shielding for the transfer assembly to reduce radiation fields to within acceptable levels during removal and packaging of the HEU core and also compares the calculated and measured radiation levels. In addition to radiation safety considerations, attention was given to the possibility of inadvertent criticality during the core removal.

It was decided that radiation exposure during the HEU fuel removal and packaging should be no more than 5 mSv for whole body doses and

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125 mSv for extremity doses, within a duration not exceeding 16 h. The target dose rates were, therefore, 3.1 mSv/h on contact with the transfer shield and 0.31 mSv/h at a distance of one meter. These target dose rates are well within the ICRP dose recommendations (International Commission on Radiological Protection, 2007), and are also conservative given that the 16-h estimated duration included a 12-h sip-test that was to be performed inside the transfer shield, and during which, personnel presence in the vicinity was to be limited. Monte Carlo calculations were performed to determine the amount of shielding that would be required to bring the transfer shield contact dose rate to within the 3.1 mSv/h limit.

2. Radiation safety analysis

In order to determine the required amount of shielding, the radionuclide inventory (source term) within the core needed to be determined. This in turn required determination of the level of depletion of the HEU fuel which was simulated using the COUPLE and ORIGEN-S computer codes from the SCALE 6.1 code package (Oak Ridge National Laboratory, 2011). Validation reports for radionuclide inventory predictions with ORIGEN-S are widely available (Gauld and Litwin, 1995; Tait et al., 1995; Gauld and Litwin, 1995). The neutron flux distribution within the HEU core, in 238 energy groups, was obtained from an MCNP5 model of the JM-1 reactor (Dennis and Puig, 2014). The COUPLE code was used to collapse existing multi-group cross sections using the neutron flux spectrum from the MCNP5 transport calculations and combine the resulting neutron spectrum-weighted cross sections to produce a binary ORIGEN-S library that is specific to the JM-1 HEU reactor. As there were no reliable data available on the precise uranium isotopic composition and the impurities present in the HEU fuel, the composition used in the simulations was from the IAEA's Research Reactor Core Conversion Guidebook (International Atomic Energy Agency, 1992), which presents information on the typical composition of U-Al fuels enriched to 93.19% ^{235}U, virtually identical to the 93.18% enriched fuel used in the JM-1 reactor. Throughout its life, the JM-1 reactor operated at various power levels for varying lengths of time and replicating this operating pattern explicitly in the simulation was not only impractical, but also unnecessary since the radionuclide inventory is primarily driven by the total core burn-up. The 30-year average reactor power of 0.365 kW, estimated from the operational history of the reactor, was used to simulate continuous operation of the reactor and to estimate further depletion of the HEU core up to 42 days, the planned cooling period, before core removal. The planning of the core removal began several months in advance, so a period of projected reactor operation was included in the analysis since the short-lived radionuclides have a significant effect on the short-term radioactivity after reactor shutdown.

The results indicated that fuel burn-up was 4.212 MWd or 101.08 MWh with approximately 5.4 g or 0.65% of the ^{235}U consumed. The mass of ^{239}Pu produced was estimated to be 0.023 g. The core activity as a function of time after reactor shutdown is shown in Fig. 1.

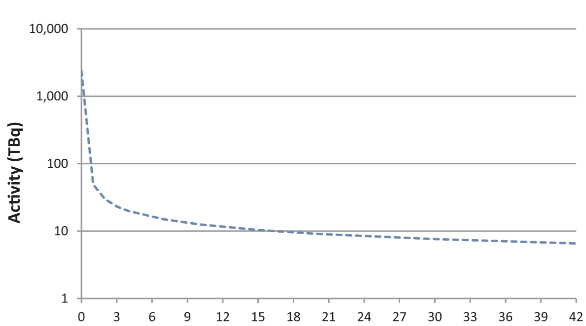

Fig. 1. Fission product activity after reactor shutdown.

Fission product activity rapidly decreased from approximately 2,394 TBq immediately after shutdown to about 49 TBq after one day and 6.57 TBq after 42 days. The fission product activity, in Bq, was also estimated using the equation (Lamarsh and Baratta, 2001):

$$3.7x10^{10}x\,A = 1.4x10^{6}P\,[t^{-0.2} - (t + T)^{-0.2}] \tag{1}$$

where A is the activity in Curies (Ci), P is the constant power in MW, T is operating time in days and t is decay period in days. For the JM-1 HEU core, after 11,531 days of operation at 0.365 kW, this equation gives an activity of 6.03 TBq after a 42 day decay period. For direct comparison, ORIGEN-S was used to simulate the operation of the reactor at constant average power for the same number of days resulting in an activity of 5.78 TBq, a difference of only 4.3% compared to that calculated using (1). The calculated particle emission rates were 4.476 $\times$ 10^{12} photons/s and 312 neutrons/s, ranging in energy from 10 keV to 12 MeV and 0.025 eV to 20 MeV respectively, 42 days after shutdown. This combined information formed the source term for the dose rate calculations.

A 3-D model of the reactor room, based on actual dimensions was created using the Monaco code from the Scale 6.1 code package (Oak Ridge National Laboratory, 2011). All 296 fuel pins, along with the central spindle and upper and lower flanges of the fuel cage were modeled. Gamma and neutron energy spectrum distributions, in 47 and 238 groups respectively, as well as emission rates were incorporated into the source definition. The source particles were sampled from the volume occupied by the fuel to account for the self-shielding of fuel pins and their cladding. All portions of the model occupied by air were modeled as voids. Built-in flux-to-dose conversion coefficients in the Monaco code were used to calculate dose rates using point detectors.

Validation of the model was performed by inserting thermoluminescent dosimeters (TLDs) into an inner irradiation site of the shutdown reactor. Two dosimeters were sequentially inserted into the irradiation site for 30 s and 60 s respectively. The average dose rate in that irradiation site was measured to be 7.02 Sv/h ± 15%. A model was created to replicate measurement conditions including the previous three days' operating pattern and decay time. The calculated dose rate in the inner irradiation site was 7.42 Sv/h ± 1.7%, a difference of approximately 5% in comparison to the measured values.

A second set of TLD measurements was performed in the same irradiation site after some urgent and unforseen full-power irradiations had to be done just before the shutdown period and the measured dose rate was 10.6 Sv/h ± 15%. As before, the model was modified to replicate these measurement conditions and the calculated dose rate was 11.2 Sv/h ± 1.8%, again only an approximate 5% difference between the calculated and measured values. There appeared to be a consistent 5% overestimation of measured dose rates.

3. Transfer shield design

A simplified diagram of the transfer shield, manufactured based on these calculations and weighing approximately 545 kg, containing the HEU core is shown in Fig. 2.

The overall design was chosen to allow the shield to be positioned just above the reactor container, bottom-load the core into the shield

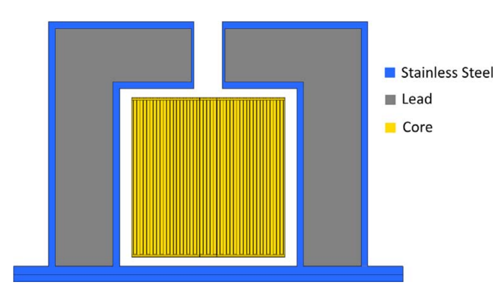

Fig. 2. Cross-sectional diagram of transfer shield containing HEU core.