Radiation Protection Dosimetry (2013), Vol. 155, No. 3, pp. 369–373
Advance Access publication 7 May 2013
doi:10.1093/rpd/nct009

COMPARISON OF MEASURED AND CALCULATED CONCRETE AND REBAR SPECIFIC ACTIVITY DURING DECOMMISSIONING OF THE DALHOUSIE SLOWPOKE-2 REACTOR

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Received July 16 2012, revised November 13 2012, accepted January 14 2013

Following the defuelling and dismantling of the Dalhousie University SLOWPOKE-2 Reactor (DUSR) in 2011, the reactor pool concrete and rebar were sampled to support the unconditional free release of the material such that the facility could be classified for unrestricted use. A detailed MCNP5 model of the critical core assembly was simulated to calculate the thermal, intermediate and fast neutron flux profile below the reactor pool floor. The neutron fluxes were used to calculate the specific activity of significant radionuclides in the concrete and rebar. The calculated specific activity and consequently the calculated neutron fluxes were validated at a number of sample locations. The calculated concrete and rebar specific activity were found to be in good agreement with the measured specific activity at the sample locations. The unrestricted use of the facility was granted through the approval of the licence to abandon the facility in August 2011.

INTRODUCTION

The SLOWPOKE-2 Reactor® is a pool type reactor with a light water (H_2O) moderated, high-enriched uranium core. The core is surrounded by a beryllium reflector assembly and is cooled by natural convection of the light water moderator. The core and reflector are installed at the bottom of a cylindrical light water-filled container, which is suspended in a below grade pool. The reactor assembly (core, beryllium reflector, control rod, instrument probes) is housed in a bolted and sealed reactor container. The reactor container can be opened for the addition of beryllium shim plates to the top reflector to maintain system excess reactivity.

The reactor container consists of a lower and upper section and the critical assembly is contained in the lower section. The upper section of the reactor container provides radiation shielding above the critical assembly. The separable upper and lower section of the reactor container enables the core, beryllium reflector, control rod, irradiation tubes and instrument lines to be removed from the reactor at the end of its useful life, while still maintain the water shielding.

The DUSR was commissioned in 1976 and was operational until 2009. The DUSR was licensed to operate at a maximum power level of 20 kWth. The DUSR had attained a cumulative burnup of 313 000 kWh over 33 y of operation. Operating experience from a similar SLOWPOKE-2 reactor that was decommissioned in 1997 indicated that the concrete and rebar located below the reactor could be free released without removal. During the sampling of the concrete and rebar, it was found that the specific activity of radionuclides at various locations below the reactor were higher than the unconditional free release criteria as defined in SOR-2000/207[(1)] and IAEA Safety Guide RS-G-1.7[(2)]. A remediation plan was developed to remove the concrete for disposal as solid radioactive waste; however, a full characterisation of the concrete below the pool and extending to a diameter of ~70 cm was required to minimise the volume of radioactive waste generated and consequently minimise damage to the pool structure.

This study quantifies the neutron flux profile below the reactor and documents the validation of various measured concrete and rebar core samples. The neutron fluxes were calculated using MCNP5[(3)]. The specific activity of various radioisotopes was calculated using the SCALE[(4)] module ORIGEN-S[(5)]. The calculated specific activities are compared with measurements.

METHODOLOGY

DUSR description

Core and beryllium reflectors

The DUSR core consists of a fuel cage and 342 element positions spaced in an equilateral triangle lattice pitch. A total of 295 fuel elements were loaded into the DUSR at the time of commissioning. The fuel cage and fuel sheath are made from ASTM type 5052 aluminium alloy. The fuel is a uranium and aluminium metal alloy. The uranium is enriched to 93 wt % ^{235}U.

The beryllium reflector system consists of annular, bottom and top reflectors and an extended life beryllium annulus, which was installed in 1986 to extend the operational life of the reactor. The top reflector consists of removable beryllium shims to compensate for ^{235}U burnup and the accumulation of reactor poisons. The annular and bottom reflector is 10.16 cm (4 in) thick and the top reflector can accommodate the same thickness of beryllium shims.

Other components

Irradiation tubes provide access to the neutron flux for sample irradiation. There are five small tubes that terminate in the beryllium annulus and five large tubes on the periphery of the beryllium annulus. Other components consist of a neutron flux detector, water thermocouple and cadmium control rod. Both the detector and thermocouple are inserted through aluminium guide tubes from the reactor container top plate. The cadmium control rod moves in the centre of the fuel cage and is guided by a tube in the centre of the tray containing the upper removable beryllium reflector plates. A general view of the critical assembly in the lower reactor container is shown in Figure 1.

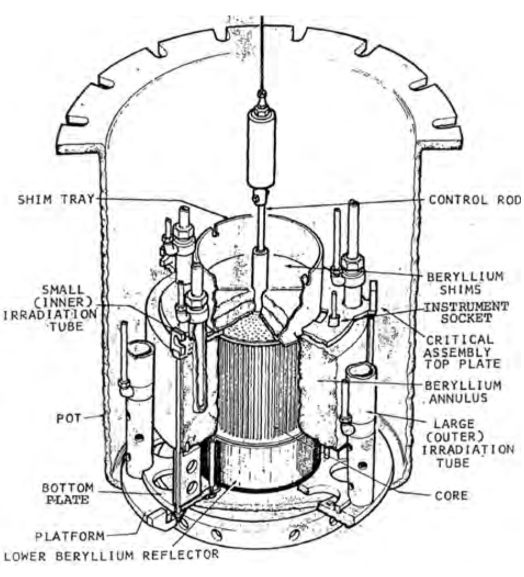

Figure 1. Dalhousie University Critical Assembly.

Neutron flux calculations

The DUSR core and beryllium reflector assembly was modelled in detail using MCNP version 5. The irradiation tubes, flux detector and thermocouple were not modelled and are expected to have a negligible effect on the calculated fluxes below the reactor. The reactor pool floor was modelled ~66 cm from the centre of the fuel cage. The Evaluated Nuclear Data File B-VI continuous energy cross section library that is distributed with MCNP5 was used for the neutron transport calculations. Thermal neutron scattering cross sections are also used to account for the bound effects of nuclei in the light water moderator and beryllium reflectors.

MCNP5 was run in criticality mode assuming a fresh core loading (no fission products). The volume-averaged neutron flux was calculated in a cylindrical scoring cell slightly above the pool surface to validate the calculated neutron flux against the measured neutron flux from a similar demonstration reactor. The volume-averaged neutron flux was also calculated below the reactor pool floor to a depth of 22.86 cm into the concrete over a diameter of 70 cm. The fluxes extending beyond this point can be extrapolated if required to avoid the use of significant computational resources and modelling techniques involving variance reduction to obtain statistically acceptable fluxes. The F4 tally option was used to calculate the neutron track-length per unit volume in each scoring cell in three neutron energy groups: thermal (<0.625 eV), intermediate (>0.625 eV <1 MeV) and fast (>1 MeV). The scoring cells were modelled as annular regions of concrete that are concentric with the reactor axis. The statistical uncertainty in the calculated neutron fluxes are all below 5 %, except for one scoring cell located at the periphery of the calculation model. This particular scoring cell had a slightly higher relative error (5.9 %), but is still considered reliable based on the general profile of the neutron flux in surrounding scoring cells.

Specific activity calculations

The chemical composition of the pool concrete was measured and found to contain 6.4 ppm of cobalt, 403 ppm of manganese, 1.24 ppm uranium and 35 ppm of zinc. The iron content was inferred from a similar SLOWPOKE-2 reactor commissioned around the same time and decommissioned in 1997. The rebar chemical composition was measured as well and found to contain 94.4 wt % iron, 50.5 ppm of cobalt and <0.05 ppm of europium. These impurities under activation in a neutron flux produce the main isotopes of interest: ^{60}Co, ^{54}Mn, ^{55}Fe, ^{152}Eu and ^{154}Eu. These isotopes represent more than 95 % of the total activity in the concrete.

The specific activity of radionuclides in the concrete and rebar were calculated using the SCALE6 module ORIGEN-S. ORIGEN-S was run in the flux-irradiation mode using the thermal, intermediate and fast fluxes calculated using MCNP5 for the full-core reactor simulation. The fluxes used in ORIGEN-S were normalised to the actual power history in each individual year of operation over 33 y of cumulative reactor operation. The reactor was operated at a power level of 16 kWth from 1976 to 1978, 8 kWth from 1978 to 2001, 4 kWth from 2001 to 2008 and 2 kWth until 2009.

The specific activity was calculated assuming 530 d of decay prior to decommissioning activities. Concrete and rebar samples were simulated at the locations given in Table 1. A typical concrete sample has a diameter of 10.16 cm and depth of 15.24 cm. The flux was averaged over a depth of 12.7 cm in the simulations. The fluxes used in the rebar assessment are the volume-averaged neutron flux calculated between a depth of 15.24 and 17.78 cm. Sample C0 is located directly below the reactor. C0–1 is identical to C0 except that the sample depth extends from 15.24 to 22.86 cm. The volume-averaged neutron flux between a depth of 15.24 and 22.86 cm was used to simulate sample C0–1.

Table 1. Concrete and rebar sample locations.

Sample ID	Distance from reactor axis (cm)	Sample depth (cm)
C0	On axis	15.24
C0–1	On axis	22.86
C1	~20	15.24
R1	43.18	15.24
R2	43.18	15.24
R3	30.48	15.24

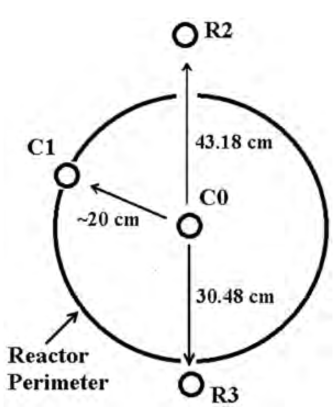

Figure 2. The concrete and rebar sample ID and distance to the reactor core axis.

Samples C1, R1, R2 and R3 are illustrated in Figure 2. Note that other locations were sampled on the pool floor and side walls; however, only those locations identified in Table 1 showed specific activity levels that were in excess of the unconditional free-release limits.

RESULTS AND DISCUSSION

Neutron fluxes

The thermal neutron flux measured at the surface of the reactor pool floor of a similar demonstration reactor operating at 16 kWth was reported as ~6E+07 n cm^{-2} s^{-1}. The thermal neutron flux calculated using the MCNP5 full-core model is 7.25E+07 n cm^{-2} s^{-1}. This is good agreement provided the calculated thermal neutron flux is volume averaged over a 5.08 cm diameter and 2.54 thick region of water above the pool floor; thus, the fluxes will be slightly higher. The thermal (<0.625 eV), intermediate (>0.625 eV <1.0 MeV) and fast (>1 MeV) neutron flux profile in the reactor pool floor concrete is given in Tables 2–4, respectively. The calculated neutron fluxes have a statistical uncertainty of <5 %. The neutron fluxes are applicable to a SLOWPOKE-2 reactor operating at a power level of 16 kWth.

Comparison of calculations with measurements

The specific activity measurements were performed by Atomic Energy of Canada Limited using an ISOCS portable High-Purity Germanium detector (HPGe). The measurement uncertainty was found to vary from 10 to 20 %. The measured and calculated specific activity of rebar and concrete is shown in Table 5. The calculated values are shown in parentheses. A comparison of the calculated and measured specific activity in concrete shows that both are in good agreement for samples located directly beneath the reactor (C0 and C1). The calculated specific activities at R1, R2 and R3 show reasonable agreement for ^{152}Eu, ^{54}Mn, ^{60}Co and ^{154}Eu (within 50 % of the calculated values). The ^{54}Mn specific activity calculated for rebar is within 20 % of the measured values. The calculated ^{152}Eu specific activity in rebar is within 40 % of the measured specific activity. The calculated ^{60}Co shows reasonable agreement as well, varying from 30 % at C0 to 46 % at R3.

There are statistical uncertainties in the calculations, which include the uncertainty in the measured elements of the concrete and rebar, uncertainty in the neutron flux calculations and neutron transport cross section libraries. The ORIGEN-S activation cross section libraries are also based on a PWR neutron spectrum, which may introduce additional uncertainty in the calculated specific activities. Given these

Table 2. Thermal neutron flux (n cm^{-2} s^{-1}) as a function of depth in concrete and distance from the reactor axis at maximum power (16 kWth).

Depth (cm)	Radial distance from reactor axis (cm)							
	0–5	5–10	10–20	20–30	30–40	40–50	50–60	60–70
2.54	2.33E+07	2.26E+07	1.89E+07	1.30E+07	7.50E+06	3.97E+06	1.91E+06	8.33E+05
5.08	1.87E+07	1.83E+07	1.51E+07	1.04E+07	6.29E+06	3.42E+06	1.65E+06	7.41E+05
7.62	1.47E+07	1.43E+07	1.23E+07	8.53E+06	5.20E+06	2.90E+06	1.46E+06	6.70E+05
10.16	1.22E+07	1.17E+07	9.90E+06	6.93E+06	4.32E+06	2.43E+06	1.26E+06	5.65E+05
12.70	1.00E+07	8.89E+06	7.66E+06	5.45E+06	3.16E+06	1.88E+06	1.06E+06	4.82E+05
15.24	7.80E+06	6.85E+06	5.74E+06	4.35E+06	2.73E+06	1.58E+06	8.12E+05	3.88E+05
17.78	5.32E+06	5.17E+06	4.25E+06	3.23E+06	2.06E+06	1.16E+06	6.31E+05	3.10E+05
20.32	3.68E+06	3.44E+06	2.87E+06	2.23E+06	1.41E+06	8.11E+05	4.43E+05	2.17E+05
22.86	2.05E+06	1.84E+06	1.56E+06	1.19E+06	7.58E+05	4.69E+05	2.53E+05	1.14E+05

Table 3. Intermediate neutron flux (n cm^{-2} s^{-1}) as a function of depth in concrete and distance from the reactor axis at maximum power (16 kWth).

Depth (cm)	Radial distance from reactor axis (cm)							
	0–5	5–10	10–20	20–30	30–40	40–50	50–60	60–70
2.54	1.16E+07	1.13E+07	9.52E+06	6.44E+06	3.67E+06	1.96E+06	9.34E+05	3.96E+05
5.08	1.01E+07	9.92E+06	8.54E+06	6.02E+06	3.41E+06	1.91E+06	9.28E+05	4.05E+05
7.62	9.00E+06	8.58E+06	7.44E+06	5.35E+06	3.18E+06	1.78E+06	9.06E+05	3.83E+05
10.16	7.31E+06	7.35E+06	6.38E+06	4.50E+06	2.84E+06	1.58E+06	8.17E+05	3.53E+05
12.70	6.16E+06	5.94E+06	5.33E+06	3.95E+06	2.46E+06	1.40E+06	7.30E+05	3.35E+05
15.24	5.36E+06	5.01E+06	4.33E+06	3.27E+06	2.05E+06	1.19E+06	6.32E+05	3.09E+05
17.78	3.79E+06	4.01E+06	3.40E+06	2.59E+06	1.61E+06	9.18E+05	5.07E+05	2.59E+05
20.32	3.16E+06	2.65E+06	2.51E+06	1.88E+06	1.18E+06	6.81E+05	3.85E+05	1.89E+05
22.86	1.78E+06	1.52E+06	1.41E+06	1.12E+06	7.28E+05	4.19E+05	2.45E+05	1.10E+05

Table 4. Fast neutron flux (n cm^{-2} s^{-1}) as a function of depth in concrete and distance from the reactor axis at maximum power (16 kWth).

Depth (cm)	Radial distance from reactor axis (cm)							
	0–5	5–10	10–20	20–30	30–40	40–50	50–60	60–70
2.54	8.29E+06	7.73E+06	6.56E+06	4.41E+06	2.53E+06	1.32E+06	6.42E+05	2.95E+05
5.08	6.64E+06	6.57E+06	5.51E+06	3.80E+06	2.28E+06	1.19E+06	5.97E+05	2.54E+05
7.62	5.58E+06	5.31E+06	4.32E+06	3.16E+06	1.88E+06	1.06E+06	5.22E+05	2.39E+05
10.16	4.36E+06	4.29E+06	3.49E+06	2.63E+06	1.55E+06	8.70E+05	4.58E+05	2.02E+05
12.70	3.37E+06	3.27E+06	2.89E+06	2.17E+06	1.29E+06	7.42E+05	4.16E+05	1.83E+05
15.24	2.77E+06	2.59E+06	2.31E+06	1.73E+06	1.06E+06	6.11E+05	3.48E+05	1.63E+05
17.78	1.99E+06	1.96E+06	1.75E+06	1.35E+06	8.76E+05	5.21E+05	2.79E+05	1.39E+05
20.32	1.52E+06	1.53E+06	1.38E+06	9.84E+05	7.17E+05	4.02E+05	2.22E+05	1.13E+05
22.86	1.11E+06	1.18E+06	1.03E+06	7.23E+05	5.22E+05	3.04E+05	1.69E+05	7.99E+04

uncertainties, the comparison does show that the calculations validate the full-core representation of the SLOWPOKE-2 reactor using MCNP5, the calculated neutron fluxes and the ORIGEN-S calculations used to predict the specific activity of various radioisotopes in the concrete and rebar. Since the methodology was validated, the spatial profile of additional isotopes that were not directly measured was confidently predicted using ORIGEN-S as well. These isotopes include ^{55}Fe, ^{59}Fe, ^{59}Ni and others.

Table 5. Comparison of measured and calculated specific activity (Bq g^{-1}) for the Dalhousie University SLOWPOKE-2 reactor.

Sample ID	Specific activity (Bq g^{-1})			
	^{54}Mn	^{60}Co	^{152}Eu	^{154}Eu
Concrete				
C0	0.085 (0.07)	0.87 (0.96)	4.25 (3.78)	0.24 (0.24)
C0–1	<0.01 (0.017)	0.16 (0.21)	0.86 (0.83)	<0.026 (0.05)
C1	<0.025 (0.045)	0.49 (0.66)	2.80 (2.56)	0.19 (0.16)
R1	<0.010 (0.017)	0.13 (0.26)	0.71 (0.98)	<0.03 (0.06)
R2	<0.010 (0.017)	0.11 (0.26)	0.59 (0.98)	<0.03 (0.06)
R3	<0.030 (0.045)	0.31 (0.53)	1.67 (2.07)	<0.10 (0.13)
Rebar				
C0	0.92 (0.95)	1.80 (3.06)	<0.072 (0.11)	—
R1	0.25 (0.31)	0.47 (0.67)	0.020 (0.023)	—
R2	0.20 (0.31)	0.38 (0.67)	0.013 (0.023)	—
R3	0.53 (0.62)	1.00 (1.86)	0.039 (0.065)	—

CONCLUSIONS

The neutron flux profile below the reactor pool floor was calculated using MCNP5, which was used as an input to ORIGEN-S to calculate the specific activity for validation against concrete and rebar sample measurements. The calculated specific activity of the concrete and rebar was compared against measurement of select radioisotopes and the calculations are in reasonable agreement with measured values. The specific activity measurements validate the calculation methodology used in this analysis.

The results of this analysis may be used in future SLOWPOKE-2 decommissioning campaigns to accurately predict radioisotope concentrations in concrete and rebar through the use of the detailed spatial neutron flux profile presented here. The flux profile should be prorated to the power history of the specific SLOWPOKE-2 reactor. This work may then be helpful in preparing a concrete and rebar sampling plan, input to a waste management plan and support in the application for a licence to abandon a Canadian SLOWPOKE-2 facility.

FUNDING

Funding for this work was provided by Atomic Energy of Canada Limited.

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