Arab J. Nucl. Sci. Appl., Vol. 56, 2, 1-8 (2023)

Paper presented in the 10th African Conference on Research Reactor Safety, Operation & Utilization, Cairo (Egypt) 27 to 29 Nov. 2022, organized by the International Atomic Energy Agency in cooperation with the Egyptian Atomic Energy Authority

ESNSA

ISSN 1110-0451

Arab Journal of Nuclear Sciences and Applications

Web site: ajnsa.journals.ekb.eg

IAEA

Radiological Characterization and Safety during Decommissioning of a Nuclear Reactor

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ARTICLE INFO

Article history:

Received: 24th Oct. 2022

Accepted: 26th Dec. 2022

Keywords:

Decommissioning;

Research Reactor;

Safety;

MCNPX;

Inventory;

Activity.

ABSTRACT

As part of the decommissioning plan of a research reactor, one of the stages is the transport of the spent fuel from the reactor site to dry storage or to country of origin. Consequently, the radiological characterization for spent nuclear fuel is necessary for shielding calculation and dose rate estimation on the transport cask surface. It is the basis for radiation protection and safety of personnel, public and the environment. It support the identification of contamination and assessment of potential risks. Most of the comprehensive characterization activities are conducted during the transition stage in preparation for decontamination and dismantling activities. Several methods were utilized for spent fuel inventory calculation and activation products in structure materials of a nuclear reactor. In this paper concentrations and time- dependent activities of the source term of OPAL research reactor spent fuel will be calculated using MCNPX 2.7 Monte Carlo code. Calculations based on operational irradiation history in cycle 7, initial enrichment, discharge BU and neutron cross-section data library. Nuclides composition can be changed during cooling times as different decay schemes lead to production and destruction of nuclides. Also, radiation protection and safety management for the workers, public and for the environment in case of normal and emergency exposure during decommissioning activities will be evaluated based on the international basic safety standards.

1. INTRODUCTION

In decommissioning activities, radiation protection plays an important role that depends on several aspects such as:

- The strategy of decommissioning
- Decommissioning activities classification
- Techniques selection of decontamination and dismantling
- Waste management concept, especially concerning treatment of radioactive material towards clearance or disposal, and options for logistics

The radiological characterization is fulfilled by the following items:

- Identification of isotopic composition, contamination level in spent fuel, structures, and components
- Identification of the nature and extent of remedial actions and decontamination methods
- Facilitating the preparation for decommissioning

Decommissioning project is divided into several phases to help personnel with learning and integrating their experience during work. This method assists with reduction of time while supporting safety and radiation protection, also, with additional information of radiological inspections, it contributes to the waste management process as it is required for waste characterization [1-7]. As a learnt lessons, it was found that to improve decommissioning project schedule, the removal of large components should be applied to ex-situ cutting as an alternative of in-situ cutting; that approach consist of [8-9]:

(a) component separation in the facility,

(b) shutting down all openings,

(c) Component is taken out the nuclear facility’s original location.

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DOI: 10.21608/AJNSA.2022.170609.1653

Decontamination and removed component can be made outside the site by a service provider. Based upon radionuclides inventory of a component and techniques applied the decontamination and cutting can result in materials released from regulatory control [10-11].

In this paper, radiological characterization of OPAL research reactor 7th operational cycle spent fuel has been investigated using Monte Carlo code (MCNPX 2.7) to study radionuclides concentrations and time- dependent activities of the source term of OPAL spent fuel. The research reactor benchmarking data base including the operating history of OPAL was published in IAEA technical report series No 480 (Rev.1) [16,18].

2. Radiological characterization for decommissioning

2.1. Measurement and sampling aspects

For radiological characterization, the following measurements applied for each component [11- 14],

- Dose rate measurements
- Surface contamination monitors
- Gamma spectrometry
- Sampling and subsequent laboratory analysis in the presence of alpha and beta emitters and gamma radiation requires
- Overall contamination intensity determined by in situ gamma spectrometry or dose rate.

Localized contamination is determined by means of specified methods with surface contamination detectors, collimated in situ gamma spectrometry, or by swiping or substance samples. After that, it is assessed using straightforward statistical techniques like calculating the mean and variance.

That approach is significantly promising for decreasing the total decommissioning prices, via determining the average contamination level, facility area that help with deciding the necessity for decontamination.

2.2. Use of data management tools

Nowadays, there are several methods for managing data that is applied during radiological characterization and provide the data visualization on 2D/3D models, hence permitting a simple assessment of the radiological characterization progress such as;

- All data about exact position, and the responsible person who made the measurement are stored
- Data quality assurance,
- Updating the radiological data of radioactive decay
- Collection of a set of data belonging to one entity and sharing a common background in terms of operational and contamination history.
- Statistical analysis of a set of data in relation to distinctive characteristics.
- Formation of nuclide vectors and radiological signatures using established techniques
- Documentation and data archiving.

3. Radiation protection and radiological characterization

According to international basic safety standards, radiation protection is important step to consider through the preparation and execution of decommissioning projects. Several aspects affecting the workforce, affect radiation protection. Based on decommissioning plans that take into consideration the circumstances of each research reactor.

First stage of a decommissioning project requires radiological characterization of radioactive component and spent fuel to ensure the radioactive risks are identified in the detailed work. Hence, nuclear facility project strategy it is a necessity.

Radiological inspections is needed after the completion of decommissioning project to establish the expected final end state has been achieved and fulfilled the regulatory conditions.

Difficulties like these have so far been successfully addressed, however, it is beneficial to increase the experience sharing in what way to conduct radiological characterizations and improve radiological inspections, as better worldwide experience exchange during decommissioning.

Engineering controls is the preferred method for maintaining exposures to radiation according to ALARA. That include: (a) Access control, (b) decrease exposure time, (c) increase distance between the radioactive source and personnel, (d) usage of specialized Personnel Protective Equipment (PPE) and respiratory protection equipment, (e) Confinement/ containment structures, (f) controlled ventilation, (g) shielding. That efficiency of controls is evaluated by area radiation monitoring which is measured by direct dose rate, surface contamination and airborne radioactivity [6].

The level of occupational radiation exposure from external sources will be determined using personnel monitoring equipment (TLD). The amount of radionuclides in air measured via air samples, or from air monitoring and bioassay, will be used to evaluate an individual's occupational exposure to radioactive materials contained inside them.

4. Radiological Accident Management during Decommissioning

The radiological accident during decommissioning is considered during dismantling or decontamination and evaluated through examining the radioactive inventory, reviewing planned activities and any possible activity that may lead to a radioactive material release. The highest accident potential is a fire during decommissioning activities.

The effective dose to personnel is calculated under the assumption that the total activity contained in the decommissioning place is involved in the fire. The committed effective dose to an individual due to atmospheric dispersion should be lower than the effective dose constraints of 10 mSv for the general public [3].

5. Case Study: Radiological Characterization OPAL Spent Fuel

5.1. Reactor Description

The Open Pool Australian Light water "OPAL" research reactor is 20 MW thermal power. The reactor core configure of 16 fuel assemblies, each fuel assembly contains 21 fuel plates with dimensions 81.5mm x 81.5mm. Each fuel plate consists of Aluminum clad plate of low-enriched uranium Silicide dispersed into an Aluminum matrix. Reactor power is managed by five control rods, and it also facilitate the shutdown [16-18].

The reactor operates in cycle seven about 39 days followed by refueling outage to remove three spent fuel assemblies and replace them with fresh fuel assemblies. The uranium loading is 39.1 KgU, the assembly average discharge burn-up is 49.1 % U-235, and the maximum thermal neutron flux is 3 x 1014 n/cm^2.s, OPAL operational data is presented at Tables 1 and 2 and core configuration for cycle 7 is presented in Figure 1-3.

Table (1): OPAL Technical data [16]

Reactor Type		Open Pool
Nominal Power		20MW$_{th}$
Fuel type		Plate, U_3Si_2-Al dispersion in Al clads parallel plates
No. of plates per assembly		21
Enrichment		19.8wt%
Fuel density	**Type 1**	4.3530 g/cm^3
	Type 2	5.7187 g/cm^3
	Standard Fuel	6.482g/cm^3
Fuel Assembly (FA) dimensions		81.5 mm x 81.5 mm
Active height		615 mm
Meat width		65mm
Side plate dimensions		5.0 x80.5 x890 mm
Coolant (type and flow direction)		Light water, upward flow
Coolant gap		2.45 mm
Moderator		Light water
Reflector		Heavy Water

Table (2): Cycle 7 Operational data of Critical and regulating plate position (%) [18]

EFPD[1]	Power (MW)	CR1	CR2	CR3	CR4	CR5
0	0.00	85.07	23.47	23.10	84.97	49.90
0.58	0.69	85.06	23.47	23.10	85.17	67.54
0.92	0.00	85.05	40.06	40.02	85.06	23.31
1.02	2.41	85.05	40.06	40.02	84.91	22.10
1.08	0.00	85.00	40.00	39.91	85.01	25.41
1.80	5.81	85.00	40.00	39.91	85.01	66.20
1.92	0.00	85.04	45.99	55.02	85.06	62.62
2.79	9.99	85.04	70.60	69.04	85.06	20.09
2.83	0.00	85.00	84.23	84.75	84.96	35.21
2.96	14.36	85.00	62.32	63.95	84.96	21.01
3.00	0.00	84.98	78.05	79.87	85.07	22.34
4.04	19.26	84.98	77.54	81.56	84.93	21.00
5.08	18.74	84.98	84.08	84.55	84.93	21.31
6.13	18.80	84.98	84.08	84.55	84.93	28.33
7.51	18.78	84.98	84.08	84.55	84.93	31.74
9.42	0.00	85.02	85.03	84.87	84.97	26.59
9.92	1.10	85.02	62.12	64.18	84.97	21.03
9.96	8.26	85.02	57.27	62.72	84.97	21.03
10.21	14.76	85.02	72.65	63.33	84.97	21.03
11.29	17.94	85.02	83.96	84.05	84.97	30.29
13.33	0.00	85.08	75.03	75.49	85.01	20.30
13.38	8.63	85.08	65.59	64.42	85.01	20.30
14.38	19.19	85.08	84.01	84.00	85.01	31.92
14.63	14.49	85.08	84.01	84.00	85.01	33.08
17.32	0.00	84.97	58.67	58.16	85.12	20.06
18.13	18.73	84.97	83.99	84.09	85.12	29.85
19.08	18.75	84.97	83.99	84.09	85.12	37.51
20.17	18.75	84.97	83.99	84.09	85.12	39.13
21.25	18.75	84.97	83.99	84.09	85.12	40.44
22.42	18.77	84.97	83.99	84.09	85.12	42.00
23.27	18.76	84.97	83.99	84.09	85.12	42.86
25.29	0.00	85.17	85.13	84.90	85.05	29.14
25.33	13.52	85.17	70.97	71.01	85.05	20.12
26.42	18.97	85.17	83.91	83.97	84.91	40.92
27.58	18.93	85.17	84.06	83.97	84.91	42.97
28.75	18.99	85.17	84.06	83.97	84.91	43.08
29.92	19.54	85.17	84.06	83.97	84.91	43.19
31.25	19.60	85.17	84.06	84.11	84.91	43.10
32.46	19.59	85.17	84.06	84.11	84.91	43.19
33.71	19.55	85.17	84.06	84.11	84.91	43.57
34.92	19.43	85.17	84.06	84.11	84.91	44.20
36.13	19.37	85.17	84.06	84.11	84.91	44.95
37.46	19.40	85.17	84.06	84.11	84.91	46.73
38.83	18.70	85.17	84.06	84.11	84.91	49.03

EFPD: Effective Full Power Days [1]

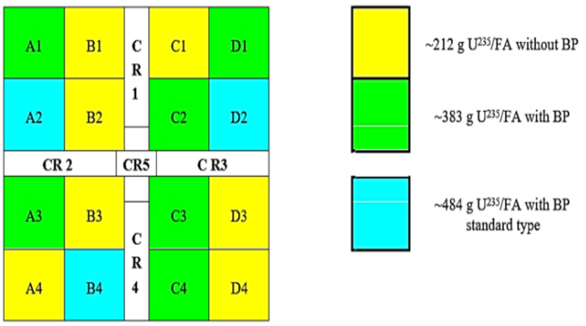

Fig. (1): FA Locations, CRPs Labels and Arrangement of Fuel Types for Cycle 7 [16]

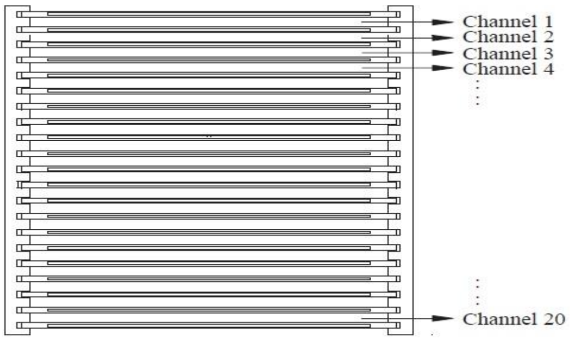

Fig. (2): OPAL Fuel Assembly

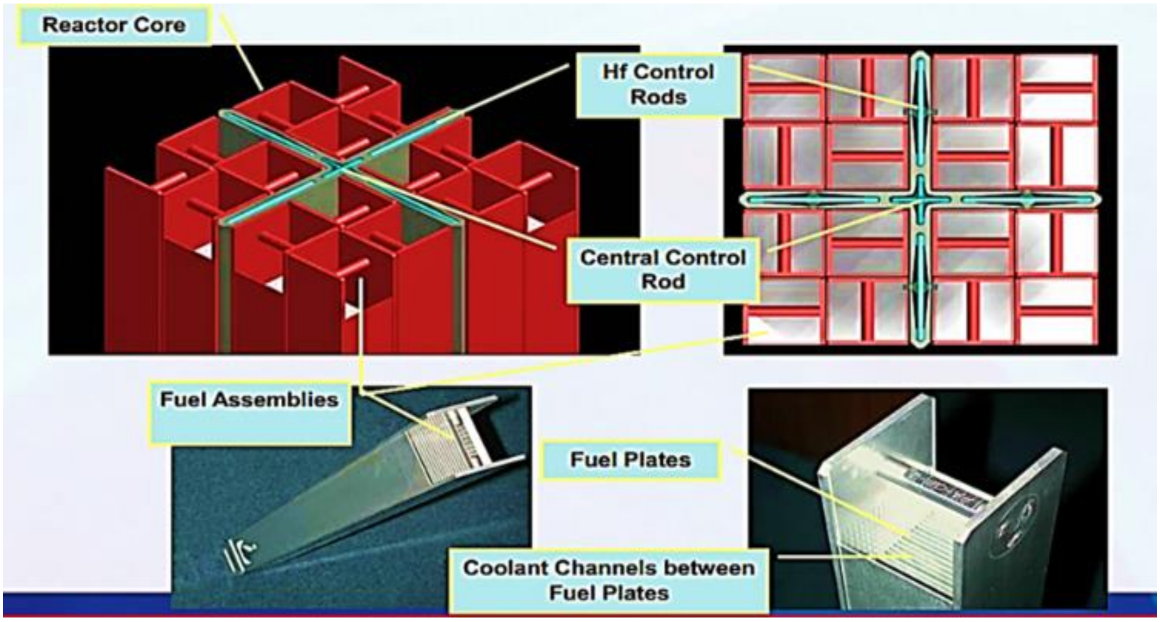

Fig. (1): Reactor Core with Five Control Rods and FAs Orientations [17]

5.2. MCNPX model of OPAL core

OPAL core was modeled using MCNPX code to perform depletion calculation, the model configurations is shown in Figures 4 and 5. MCNPX designed to track each neutron over broad ranges of energies 10^{-11} MeV to 20 MeV for all isotopes. Depletion equation that used to describe isotope decay rate is

$$\frac{\partial N_A(\vec{r},t)}{\partial t} = -N_A(\vec{r},t)\left(\lambda_A + \sum_i \sigma_a(E)\Phi_i(\vec{r},E,t)\right) + N_B(\vec{r},t)\lambda_B + N_C(\vec{r},t)\left(\sum_i \sigma_\gamma(E)\Phi_i(\vec{r},E,t)\right)$$

Where, N_A, N_B and N_C are the atomic densities (atom/cm^3) for isotope A, precursor isotope (B) by decay and precursor isotope (C) by neutron capture.

λ_A and λ_B: decay constants,

$\sigma_a(E_i)$: microscopic absorption cross section for energy group i

$\sigma_\gamma(E_i)$: radiative capture cross section for energy group i,

$\Phi_i(\vec{r},t)$: neutron flux for energy group i;

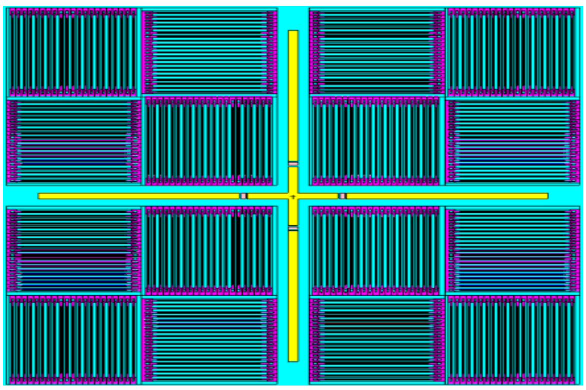

Fig. (2): OPAL Core Configuration by VISED/MCNPX

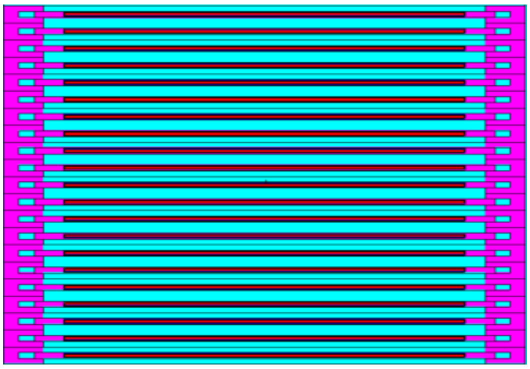

Fig. (3): Fuel Assembly by VISED/MCNPX

The characterization of spent nuclear fuel were performed to a clean fuel assembly of OPAL research reactor using MCNPX 2.7 code under the operation condition of the 7th cycle, burn-up 30GWd/MTU and at zero cooling time [15].

5.3. Model Validation

The effective multiplication factor (k_{eff}) was calculated using MCNPX code for operational cycle 7 and validated by the reactivity worth of control rods as shown in table 2.

6. RESULTS AND DISCUSSION

OPAL reactor fuel assembly-type 1 inventory and activity were calculated using MCNPX, results of the radionuclide inventory and activities are presented in Figures 5 and 6.

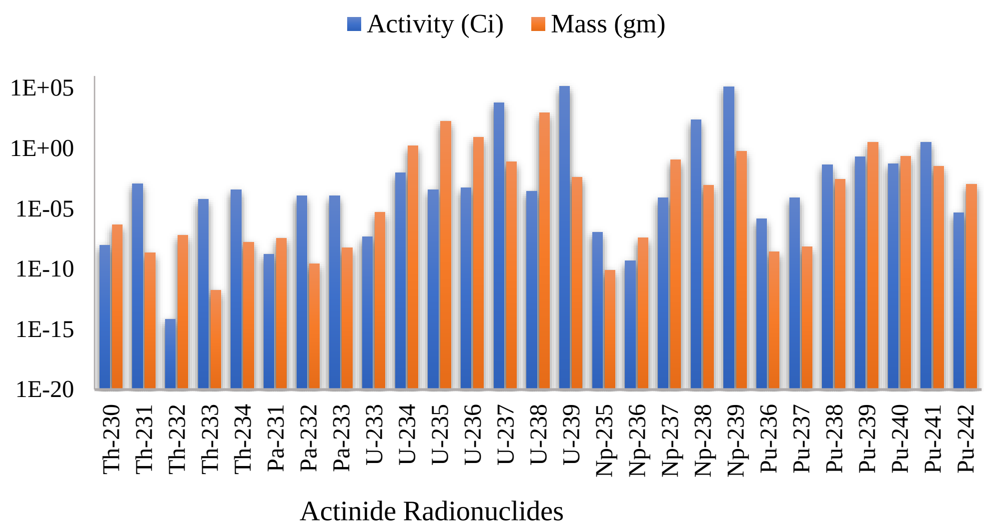

Fig. (4): Activity and Mass of Actinides

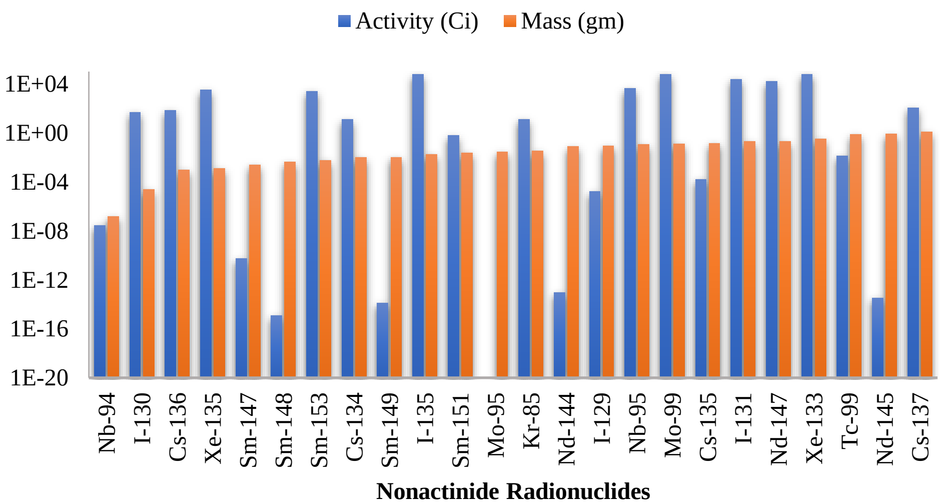

Fig. (5): Activity and Mass of Non-actinides

The results show that, the highest activity for actinides are for, U-239 and Np-239 with values, 1.34E+05Ci and 1.32E+05Ci, respectively; and for non-actinides is for, I-135 with value of 6.15E+04Ci. Also, in case of any accidental situations for the handling of OPAL spent fuel, Nobel gases as; Krypton, Iodine and Cesium will be considered, there total activities are 1.29E+01Ci, 8.58E+04Ci and1.89E+02Ci, respectively.

Also, the results of the spent nuclear fuel characterization can be used for the isotopic composition and activity calculations of the short lived and long-lived fission products. Those data can assist in the safe management, shielding design, cask type, anticipated dose rate during normal and accidental situations during the transport of the spent fuel

7. CONCLUSIONS

- Radiological characterization for spent fuel should be performed at the beginning of the decommissioning of a research reactor. The information can assist in the safe management and transport of the spent fuel, shielding design, cask type, and dose rate estimation for the safety of the workers and environment.
- As a case study, MCNPX code was used for the determination of the isotopic composition and activity calculations of short lived and long-lived fission products of OPAL spent fuel. Those data can assist in the safe management, shielding design, cask type, anticipated dose rate during normal and accidental situations during the transport of the spent fuel.

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