

Radiological Safety Evaluation of Internal Structure Activation in Transportable Micro Reactors

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1. Introduction

To address increasing regional electricity demands, research has focused on the Transportable Micro Reactor (TMR), a modular reactor concept based on Molten Salt Reactor (MSR) technology designed for enhanced mobility.[1][2] For this reason, the TMR design employs temporary external shielding during operation, thereby reducing the reliance on internal shields. This design configuration, however, internal components to significant neutron irradiation, leading to the activation of structural materials. Given operational scenarios where the reactor is relocated to subsequent demand sites post-shutdown, the absence of substantial external shielding during transport necessitates radiological safety assessment.[3]

In this study, a radiological dose evaluation was performed based on a TMR design. The analytical model assumed the complete extraction and external storage of fuel salt upon reactor shutdown to isolate the effects of fission product decay. Consequently, the analysis focused exclusively on the gamma dose rates resulting from the activation of SS316L-based internal structures. For activation and dose rate evaluation, a hybrid computational framework was employed: the deterministic activation code ORIGEN-S was used to characterize the induced radioactivity and source term strength within the structural components, while the Monte Carlo N-Particle (MCNP) transport code was utilized to simulate the neutron flux and calculate the effective dose rates at the transport container surface.

2. Methodology and computational analysis

The computational assessment of induced radioactivity and subsequent dose rates is implemented through simulation framework. This methodology systematically segments the analysis into the characterization of the neutron flux map and the subsequent isotopic activation analysis. In the initial phase, the three-dimensional transport of neutrons within the reactor core and surrounding structures is modeled using the MCNP code to determine neutron flux.

Following the flux characterization, the SCALE-COUPLE module is utilized to process cross-section libraries, which are then integrated into ORIGEN for activation calculations. This step facilitates the derivation of the time-dependent gamma emission

spectrum and source term strength from the irradiated structural materials. The final stage of the evaluation involves a secondary MCNP transport simulation, wherein the derived gamma spectrum is defined as the fixed source term to quantify the dose distribution and radiological impact at specified surface and points.

2.1 Neutron flux characterization

The MCNP code, developed by Los Alamos National Laboratory, was utilized as the primary stochastic tool for particle transport and criticality analysis.[4] To determine the neutron flux distribution during reactor operation, a three-dimensional model of the reactor was developed. While a mesh-tally approach is conventional for mapping flux distributions, this study employed a cell-averaged flux methodology to optimize computational efficiency and ensure statistical convergence across the activation zones. In the MCNP geometry configuration for the TMR, specific material assumptions were applied to represent the complex internal and external reactor environments. It was assumed that the majority of the internal structures are encapsulated in canning fabricated from SS316L. Furthermore, the outer containment was also modeled as being composed of SS316L.

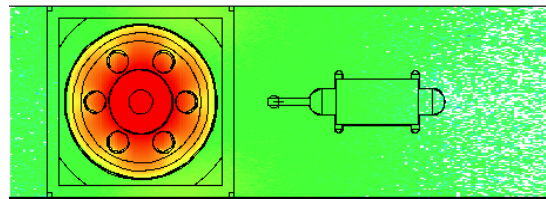


Fig 1. Neutron flux mesh map of TMR during operating (position: Z=0)

Figure 1 illustrates the neutron mesh flux map of TMR during operating, which was discretized with a spatial resolution of 1 cm. Specific neutron flux values were derived using the F4 tally, which calculates the track-length estimate of the cell-averaged flux. To normalize the tally results to the actual operating conditions, appropriate scaling factors based on the reactor's thermal power and the number of source particles were applied.

The TMR model is designed with configuration of 1.5 MWth for an extended 20-year lifespan. For the activation analysis, 64 distinct cells composed of SS316L were designated as evaluation points. The

spatially averaged neutron flux for each of these cells serves as the fundamental input for the subsequent isotopic depletion and activation calculations.

Table I. Calculated neutron flux for each cell

1.5 MWth, 20 Year	
Cell Number	Neutron Flux (#/cm ²)
84	2.89E+09
94	2.88E+09
47	4.52E+09
77	4.72E+09
...	...
849	2.49E+08
852	2.60E+08
855	2.69E+08
858	2.12E+08

Table 1 presents the normalized neutron flux values calculated for each designated cell. The neutron flux was calculated based on the initial fresh fuel conditions, thereby introducing a deliberate conservatism into the results. These data sets serve as the fundamental input parameters for the preparation of the ORIGEN.

2.2 Activation analysis and source term derivation

The SCALE (Standardized Computer Analyses for Licensing Evaluation) code system, developed by Oak Ridge National Laboratory, was employed to perform the isotopic activation analysis.[5] Within the modular architecture of SCALE, the COUPLE and ORIGEN modules were specifically utilized to characterize the material evolution under neutron irradiation.

The COUPLE module serves as a pre-processor to generate problem-dependent, one-group weighted cross-sections by collapsing the master fine-group library. Following this, ORIGEN executes the core depletion and activation calculations using the Matrix Exponential Expansion Method. This method provides a robust numerical solution to the Bateman equations, tracking the transmutation, decay, and buildup of radioisotopes over the specified irradiation period.

To ensure a conservative approach in the cross-section generation, the 238-group neutron spectrum derived from the SS316L components located on the inner side of the reflector was integrated into the COUPLE module. This selection ensures that the spectral hardening and resonance effects specific to the structural materials are accurately reflected in the activation rates. The elemental composition and mass fractions of the SS316L alloy utilized for the activation analysis are detailed in Table II.

Table II. Mass composition of SS316L

Nuclide	Composition mass ratio
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C	0.08%
Mn	2.00%
P	0.05%
S	0.03%
Si	1.00%
Cr	17.00%
Ni	12.00%
Mo	2.50%
Fe	65.35%

Based on the elemental composition defined in Table II, the isotopic activation analysis is performed by integrating the cell-specific mass, derived from its geometric volume, with the corresponding neutron flux profiles. This evaluation tracks the transmutation and decay of radionuclides over specified cooling periods to determine the time-dependent radioactivity levels.

The primary objective of this process is to characterize the gamma emission spectrum for each individual cell. By accounting for the spatial distribution of mass and the spectral variation of the neutron field, the simulation yields a discretized source term. These cell-specific gamma spectra serve as the fundamental input for subsequent transport calculations, allowing for a high-fidelity assessment of the residual radiation field following reactor shutdown.

2.3 Surface gamma dose calculation with MCNP

The final stage of the radiological dose evaluation involves determining the surface dose rates by utilizing the gamma emission spectra calculated from ORIGEN as the fixed source terms for the MCNP transport model. In this configuration, each of the 64 designated cells is assigned its corresponding discretized gamma spectrum to accurately represent the spatial distribution of the radiation source. The analysis focuses on calculating the area-averaged dose rate on the lateral surfaces of the outermost transport container.

To convert the calculated flux into a biologically relevant dose equivalent, the ICRP-74 (International Commission on Radiological Protection) flux-to-dose rate conversion factors are applied. The absolute magnitude of the final dose is derived by normalizing the simulation results with the total activity of the activated structures.

3. Evaluation of Induced Radioactivity and Surface Dose Rates

The surface dose rate evaluation was conducted based on the computational methodology described in Chapter 2. To account for the operational mobility and rapid redeployment requirements of the TMR, the reference cooling period was set to 1 day.

The analysis confirms that the primary reduction in radiation levels occurs within the first 24 hours due to the rapid decay of short-lived activation products.

Table III. Gamma Dose of 1.5MWth, 20 year condition

Surface	Dose(mSv/h)
	1 Day
1	2.59E+00
2	2.61E+00
3	3.10E-01
4	3.39E+00
Position (X, Y, Z)	Dose(mSv/h)
	1 Day
-170.1, -170.1, 0	1.23E+00
-170.1, -70.1, 0	2.51E+00
-170.1, 30.1, 0	2.80E+00
-170.1, 130.1, 0	2.14E+00
-170.1, 230.1, 0	1.62E+00
-170.1, 330.1, 0	1.25E+00
-170.1, 430.1, 0	7.38E-01

Table 3 summarize the evaluated surface dose rates for the operational configuration (1.5 MWth for 20 years). The evaluation points are categorized into four lateral surfaces of the container: Surfaces 1 and 2 represent the opposite sides of the container, while Surface 3 and Surface 4 correspond to the surfaces distal and proximal to the reactor core, respectively.

In table 3, the surface dose rates at all designated points and surfaces were found to be within the regulatory limit. This confirms that the 1.5 MWth configuration for a 20-year duration ensures radiological compliance, providing a sufficient safety margin for transport operations after a 1-day cooling period.

4. Conclusion and future work

This study evaluated the radiological safety of a TMR by assessing surface dose rates induced by the activation of internal SS316L structures. Using an MCNP-ORIGEN framework, residual radiation fields were analyzed under two distinct operational scenarios. To ensure a conservative safety margin, a constant peak neutron flux from fresh fuel conditions was assumed throughout the operation cycle.

The analysis indicates that surface dose rates stabilize after the first 24 hours post-shutdown, confirming that a 1-day cooling period is optimal for balancing mobility and safety. The 1.5 MWth (20 year) configuration achieved full compliance. This finding suggest that a low-power, long-term strategy is more favorable for the current TMR design's transportability.

Future research will utilize the 1cm-interval neutron mesh flux map developed in this work to perform fine mesh activation calculations and identify localized hot spots. Furthermore, additional studies are planned to evaluate mixed operational strategies involving various power-level combinations, providing a more comprehensive assessment of the radiological impact under realistic, flexible load-following conditions. This high-resolution approach will facilitate optimized

shielding designs to further enhance reactor safety and mobility.

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