Depletion and Decay Heat Analysis of a VHTR Core Using the McCARD/ORIGEN-2 Code

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1. Introduction

The isotopic composition of spent fuel is primarily influenced by the one-group cross-section of the nuclides that constitute the fuel during the burnup period. These one-group cross-sections depend on the neutron spectrum to which the fuel is exposed. Unlike pressurized water reactors (PWRs), boiling water reactors (BWRs), and liquid-metal-cooled reactors, the ORIGEN-2[1] code does not provide a one-group cross-section library for Very High Temperature gas-cooled Reactors (VHTRs). As a result, it is fundamentally impossible to accurately determine the radionuclide content and decay heat of spent fuel from a VHTR using ORIGEN-2 alone.

In a previous study [2], one-group cross-sections for major actinides were generated using fuel block depletion calculations with the HELIOS-1.6[3] code, and subsequently applied to ORIGEN-2 point depletion and decay heat calculations. However, due to the significant influence of the graphite reflector in VHTRs, this approach could not replace whole-core depletion calculations [4].

To address this limitation, a one-group cross-section library can be generated using a Monte Carlo code capable of performing continuous-energy, three-dimensional, core depletion simulations such as McCARD [5]. This approach can significantly enhance the accuracy of ORIGEN-2 point depletion calculations.

2. Methods and Results

2.1 PMR200 Core Design

In this study, the 200MWth prismatic VHTR(PMR200) [6] core was analyzed. The core design parameters are summarized in Table I. The radial configuration of the core is presented in Fig. 1. The fuel compacts consist of TRISO fuel particles containing 12% enriched UO_2, with a volume fraction of either 23.5% or 27.5%. Fuel blocks with 23.5% volume fraction fuel compacts are placed in the inner core, while those with a 27.5% volume fraction are located in the outer core.

The axial shuffling scheme of PMR200 is illustrated in Fig. 2, showing that the core consists of six layers. The fresh fuel batch undergoes depletion over three cycles, following a three-batch axial shuffling scheme.

Table I: PMR200 core design parameters

Parameters	Values
Thermal power (MWth)	200
UO_2 enrichment (w/o)	12.0
Kernel diameter (μm)	500
TRISO volume fraction in compact (%)	27.5/23.5
U loading (kg)	2668.2

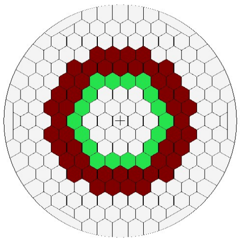

Fig. 1. Radial configuration of the PMR200 core

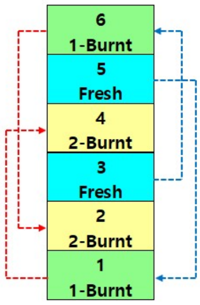

Fig. 2. Axial-only shuffling scheme for the PMR200 core

2.2 McCARD Depletion Calculation

McCARD can directly model TRISO particles while accounting for the double heterogeneity problem. However, this approach requires significant computational resources, making it impractical for depletion calculations in this study. To overcome this limitation, the reactivity equivalent physical transformation (RPT) method [7] was employed to convert the double-heterogeneous fuel compact into a conventional fuel pin model.

Fig. 3 shows the effective multiplication factor of the PMR200 core for each cycle. The cycle length is 440 days, and the equilibrium cycle is reached after 10 cycles.

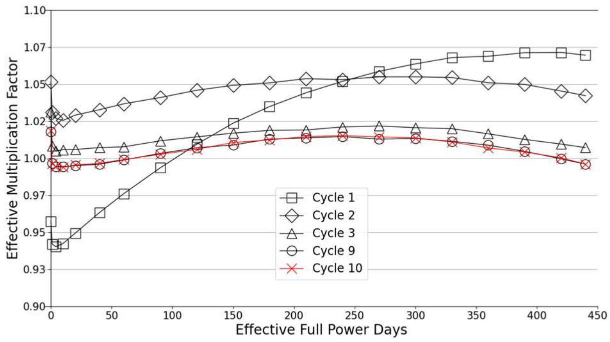

Fig. 3. Effective multiplication factor of the PMR200 core during McCARD depletion calculation

2.3 McCARD/ORIGEN-2 calculation procedures

McCARD/ORIGEN-2 depletion and decay heat calculations can be performed using two approaches. The first approach, referred to as "Core Depletion," is illustrated in Fig. 4. The Core Depletion calculation procedure involves using McCARD to determine the nuclide composition during the depletion stage, followed by ORIGEN-2 to calculate decay heat exclusively in the decay calculation stage. This approach enables ORIGEN-2 to perform decay heat calculations that fully incorporate the three-dimensional whole-core depletion results from McCARD, which employs the continuous-energy cross-section library ENDF/B-VII.

The second approach, referred to as "Point Depletion," is summarized in Fig. 5. In this method, the tally results from the McCARD depletion calculation are used to generate a new one-group cross-section library. During this process, fuel compacts are categorized into groups, and the cross-sections are averaged based on the volume and neutron flux associated with each group. This process enables the creation of a one-group cross-section library for a specific fuel group within the entire core

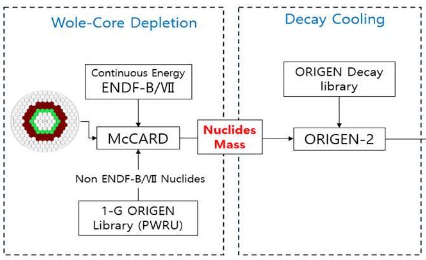

Fig. 4. Core Depletion calculation procedure

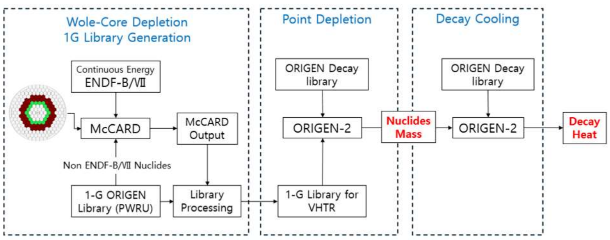

Fig. 5. Point Depletion calculation procedure

2.4 One-group cross-section library for ORIGEN-2

The one-group cross-section library in ORIGEN-2 contains both cross-section data for various nuclides and fission yield data for fission products, all represented in a one-group format for use in depletion calculations. The PWRU one-group cross-section library is used in ORIGEN-2 for PWR depletion calculations with UO_2 fuel over a burnup range of 0 to 33.0GWd/tHM.

During the McCARD depletion calculation, the PWRU library is used to account for nuclides that are not included in the continuous-energy cross-section library ENDF/B-VII. Specifically, for nuclides present in ENDF/B-VII, McCARD generates one-group cross-sections using continuous-energy calculations, whereas for nuclides absent from ENDF/B-VII, the one-group cross-sections from the PWRU library are applied.

Additionally, the yield library in McCARD stores fission yield data for each fission product in a three-energy-group format. These fission yield data are processed using a fission reaction rate-weighted average calculation before being applied to the depletion calculations.

The same approach is employed for generating the ORIGEN-2 library for PMR200 core. In the newly generated one-group cross-section library, nuclides included in ENDF/B-VII utilize one-group cross-sections obtained from McCARD continuous-energy calculations for the PMR200 whole-core model, while for nuclides not covered by ENDF/B-VII, one-group cross-sections from the PWRU library are used. Furthermore, a new ORIGEN-2 library is generated by incorporating one-group fission yield data derived from McCARD fission reaction rate tallies, ensuring more accurate fission product calculations.

2.5 Whole core depletion calculation in ORIGEN-2

The PMR200 core follows a three-batch depletion scheme, in which fuel is shifted axially after each depletion cycle. Table II presents the core states at the beginning of cycle (BOC) and at the end of cycle (EOC) for each batch during the equilibrium cycle. In the equilibrium cycle, twice-burnt batch is depleted up to 99.0GWd/tHM.

Since the ORIGEN-2 code performs point depletion calculations, which cannot model complex three-dimensional geometries, a depletion calculation must be performed for a single batch to properly represent whole-core depletion behavior.

In the equilibrium cycle, fresh fuel undergoes depletion over three cycles, and the power history of a batch during the equilibrium cycle, calculated using McCARD, is shown in Fig. 6.

Table II: Core states of the equilibrium cycle

	Batch	BOC (0days)	EOC (440Days)
Burnup (GWd/tHM)	1	0	40.5
	2	40.5	71.4
	3	71.4	99.0
	Core avg.	37.3	70.3

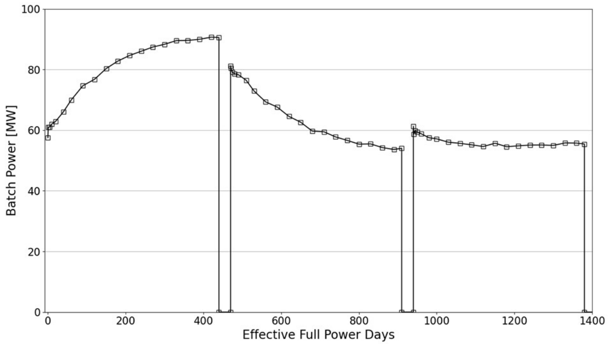

Fig. 6. Power history of a batch in an equilibrium core

2.6 Decay heat of the UO_2 fueled PMR core

To verify the library, a total of four one-group cross-section libraries were compared. The details of each library are presented in Table III. All ORIGEN-2 point depletion calculations were performed using the power history shown in Fig. 6.

After a batch was depleted up to 99.0GWd/tHM at the end of the equilibrium cycle, a 1,000-year decay calculation was performed. Fig. 7 and 8 compare the cumulative decay heat of the twice-burnt batch. The closer the results are to the Core Depletion calculations, the better the library reflects the whole-core characteristics of the PMR200.

Table III: Library descriptions

Library	Description
Batch-wise	Libraries where fuel batches with the same number of depletion cycles are grouped
Whole-Core	A Library where the entire PMR200 core fuel is treated as a single group
Block Depletion	A Library generated from depletion calculations of a single fuel block under reflective boundary conditions
PWRU	Existing library without McCARD depletion calculations

Fig. 7 compares the decay heat contribution from fission products, which remains nearly identical in most cases, as fission product decay heat is primarily influenced by the reactor's power history used in the depletion calculation. However, the PWRU library result exhibits slight deviations, as it does not account for the neutron spectrum and burnup characteristics of the VHTR.

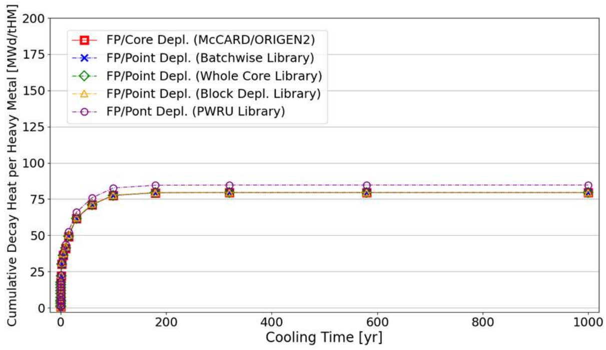

Fig. 7. Cumulative decay heat from fission products over 1,000 years

In contrast, Fig. 8 examines the actinide decay heat, where significant discrepancies appear if the library does not accurately reflect the neutron spectrum and burnup characteristics of the core.

When using the PWRU library, which does not account for any PMR200 core characteristics, or the Block Depletion library, which fails to capture the whole-core neutron spectrum, significant deviations from the Core Depletion result are observed.

Fig. 9 presents the neutron spectrum at the beginning of the cycle for both a single fuel block and the whole-core model of PMR200. In the whole-core model, the neutron spectrum tends to be softer due to the influence of the graphite reflector.

In contrast, the Batch-wise library provides the most accurate results, as it effectively incorporates the neutron spectrum and burnup characteristics of the core. The Whole-core library shows relatively larger discrepancies compared to the Batch-wise library. While it captures the whole-core neutron spectrum, it does not sufficiently reflect batch-dependent burnup variations.

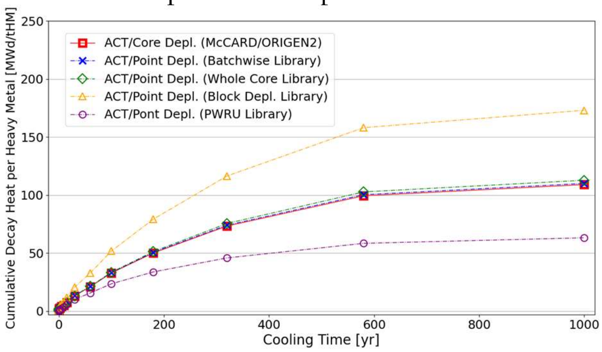

Fig. 8. Cumulative decay heat from actinides over 1,000 years

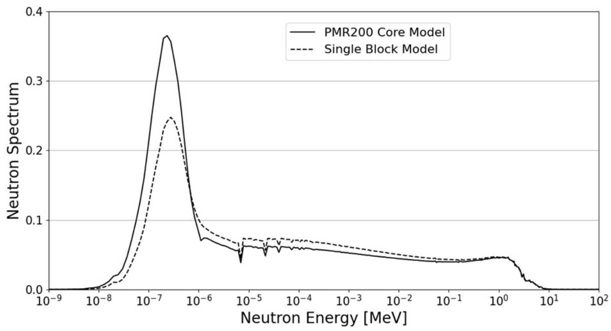

Fig. 9. Comparison of the fuel zone neutron spectrum

2.7 Nuclide mass of the UO_2 fueled PMR core

Table IV presents the nuclide inventory after the twice-burnt fuel batch reaches 99.0 GWd/tHM in the equilibrium cycle. The listed nuclides include major actinides, which account for 99.86% of the total actinide mass, as well as fission products that are considered representative source terms in VHTR fuel analysis [8].

According to the table, unlike the Block Depletion library, the Batch-wise library performs depletion calculation that closely matches the McCARD core depletion result even at high burnup for major nuclides. In particular, by incorporating fission yield data, it also enables more accurate depletion calculations for fission products.

Table IV: Comparison of mass in the fuel at the discharge burnup (99.0GWd/tHM)

Nuclide	McCARD [g/tHM]	ORIGEN2 Depletion			
		1-G Library from Batch-wise Core Depletion		1-G Library from Block Depletion	
		Mass [g/tHM]	Error [%]	Mass [g/tHM]	Error [%]
U235	3.48E+04	3.44E+04	1.1	4.18E+04	20.3
U236	1.48E+04	1.48E+04	0.2	1.45E+04	1.8
U238	8.30E+05	8.29E+05	0.0	8.13E+05	2.0
Np237	1.05E+03	1.06E+03	1.4	1.46E+03	39.4
Pu239	7.10E+03	7.20E+03	1.5	1.28E+04	79.8
Pu240	4.20E+03	4.13E+03	1.5	4.43E+03	5.5
Pu241	3.38E+03	3.38E+03	0.0	5.63E+03	66.7
Pu242	2.08E+03	2.08E+03	0.0	2.30E+03	10.8
Am241	1.08E+02	1.08E+02	0.0	2.11E+02	95.8
Xe133	1.76E+01	1.78E+01	0.8	1.78E+01	1.1
Kr85	7.62E+01	7.67E+01	0.6	7.27E+01	4.6
Kr88	9.14E-02	9.19E-02	0.5	8.47E-02	7.3
I131	1.34E+01	1.35E+01	0.9	1.38E+01	3.2
I133	3.03E+00	3.05E+00	0.8	3.06E+00	1.0
Te132	7.78E+00	7.85E+00	0.9	7.95E+00	2.2
Cs137	3.57E+03	3.59E+03	0.6	3.60E+03	0.6
Cs134	2.97E+02	3.03E+02	2.0	3.63E+02	22.0
Sr90	1.69E+03	1.70E+03	0.6	1.59E+03	6.1
Ag110m	1.66E+00	1.70E+00	2.5	2.31E+00	39.4
Ag111	6.84E-01	6.88E-01	0.6	7.73E-01	13.0
Sb125	2.06E+01	2.07E+01	0.6	2.34E+01	13.6
Ru103	7.97E+01	8.04E+01	0.9	8.49E+01	6.6
Ce144	7.52E+02	7.58E+02	0.8	7.31E+02	2.8
La140	5.70E+00	5.74E+00	0.8	5.61E+00	1.6

3. Conclusions

In this study, a one-group cross-section library was developed for ORIGEN-2 point depletion calculations of the PMR200 equilibrium cycle using McCARD core depletion calculation. The new library was applied to ORIGEN-2 depletion and decay heat calculation.

Comparison with Core Depletion results demonstrated that the Batch-wise library, unlike the Block Depletion library, accurately reflects the neutron spectrum and burnup characteristics of the whole core, even at high burnup. Additionally, incorporating fission yield data enabled accurate calculations of the fission product inventory.

These results confirm that the Point Depletion calculation procedure closely reproduces the results of the Core Depletion calculation procedure.

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