NEUTRON CROSS SECTION GENERATION OF PWR MOX FUEL ASSEMBLIES WITH SCALE AND SERPENT

Vu Thanh Mai[1*], Tran Viet Phu[2] and Pham Nhu Viet Ha[2]

[1]*Hanoi University of Science, Vietnam National University,*
334 Nguyen Trai, Thanh Xuan, Hanoi 10000, Viet Nam

[2]*Institute for Nuclear Science and Technology (INST), Vietnam Atomic Energy Institute (VINATOM),*
179 Hoang Quoc Viet str., Cau Giay dist., Hanoi 100000, Viet Nam

**E-mail: mai_vu@hus.edu.vn*

Abstract: The SCALE/TRITON code (based on deterministic method) and the Serpent 2 code (based on Monte Carlo method) were utilized in this study to prepare the group constants of the pressurized water reactor (PWR) mix-oxide (MOX) assemblies for transient analyses of PWR MOX fueled cores in normal operation and rod ejection accident condition with 3D reactor kinetics codes. The PWR MOX assemblies were modeled with TRITON and Serpent and its infinite multiplication factor versus burnup and respective neutron cross sections were calculated and compared with the available benchmark data. The comparative results generally show a good agreement between TRITON and Serpent with the benchmark data, indicating that the TRITON and Serpent models developed herein for the PWR MOX assemblies can be applied to group constant generation to be further used in transient analyses of PWR MOX fueled cores.

Keywords: PWR MOX assembly, group constants, SCALE, Serpent

I. INTRODUCTION

The utilization of recycled plutonium as mix-oxide (MOX) nuclear fuel in light water reactor (LWR) cores helps to save the natural uranium resources and reduce either the amount of weapon-grade plutonium or the plutonium amount which has to be disposed to the final storage. However, it is noted that rod ejection accident (REA), which is a consequence of mechanical failure of the control rod drive mechanism casing located on the reactor pressure vessel top and categorized as design-basis reactivity-initiated accident in pressurized water reactors (PWRs), is of particular concern for MOX fueled cores since their delayed neutron fractions are significantly smaller than those of UO2 cores [1]. The rod ejection transient can result in significant, localized perturbations of the neutronic and thermal-hydraulic core parameters, which can be difficult for reactor core simulators to predict accurately, especially in a heterogeneous MOX/UO2 fueled core. In our current research effort, we aimed to use the 3D reactor kinetics codes like PARCS [2] and NODAL3 [3] to examine thoroughly REAs in PWR MOX fueled cores. Hence, the present study was performed to generate the neutron cross sections for the PWR MOX fuel assemblies based on the OECD/NEA and U.S. NRC PWR MOX/UO2 Core Transient Benchmark [1]. The TRITON module (based on deterministic method) of the SCALE code system [4] and the Serpent 2 code (based on Monte Carlo method) [5] were utilized herein to do this generation. The infinite multiplication factor of the PWR MOX assemblies versus burnup and the respective neutron cross sections obtained with TRITON and Serpent were analyzed in relation to the available benchmark data.

II. CALCULATION METHODOLOGY

The PWR MOX assembly configuration was obtained from Ref. [1] with 17×17 fuel lattice including 264 fuel rods and 24 Wet Annular Burnable Absorber (WABA) pins. In the present study, the MOX 4.0% and MOX 4.3% assembly types were modeled and analysed with the SCALE/TRITON module (deterministic method) and Serpent 2 (Monte Carlo method). The

MOX assembly configuration and its models with TRITON and Serpent are represented in Fig. 1. The heavy metal composition in the MOX 4.0% and MOX 4.3% assemblies is shown in Table 1. The detailed assembly design parameters and material composition can be found in Ref. [1].

The two models have the same geometry in two dimensions with infinite assembly height and the same material composition. The only differences are the transport and depletion solution methods as well as the nuclear data libraries used in each code. The SCALE 252-group library and the ENDF/B-VII.0 data library [6] were used in the TRITON and Serpent calculations, respectively. The calculations were done under the hot full power condition. The infinite multiplication factor (k-inf) of the MOX assemblies versus burnup and the respective neutron cross sections obtained with TRITON and Serpent were analyzed in relation to the benchmark data obtained by the HELIOS code.

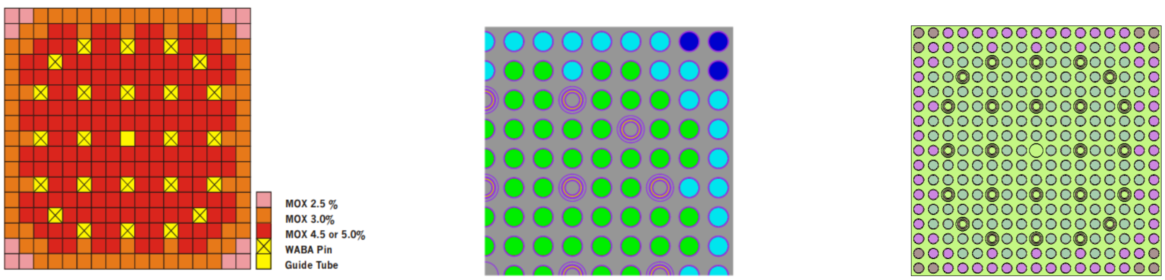

Fig. 1 MOX assembly configuration (left) and its models with TRITON - ¼ symmetry (center) and Serpent (right)

Table 1 Heavy metal (HM) composition in MOX fuel

Assembly type	Density [g/cm^3]	HM material	
MOX 4.0%	10.41	Corner zone: 2.5 wt% Pu-fissile	Uranium vector: 234/235/236/238 = 0.002/0.2/0.001/99.797 wt% Plutonium vector: 239/240/241/242 = 93.6/5.9/0.4/0.1 wt%
		Peripheral zone: 3.0 wt% Pu-fissile	
		Central zone: 4.5 wt% Pu-fissile	
MOX 4.3%	10.41	Corner zone: 2.5 wt% Pu-fissile	
		Peripheral zone: 3.0 wt% Pu-fissile	
		Central zone: 5.0 wt% Pu-fissile	

III. RESULTS AND DISCUSSION

The k-inf values of the MOX 4.0% and MOX 4.3% assemblies versus burnup obtained with TRITON and Serpent were presented in Fig. 2 in comparison with the reference benchmark data obtained with HELIOS [1]. It was shown that there is generally a good agreement between the three codes. The maximum deviations of the TRITON and Serpent results from the benchmark data were 479 pcm and 564 pcm, respectively, for the MOX 4.0% assembly. These maximum values were 588 pcm and 643 pcm for the MOX 4.3% assembly. In addition, as the fuel burns out, the TRITON results tend to overpredict the benchmark data while the Serpent results tend to underpredict the benchmark data.

The two-group neutron cross sections were also generated using TRITON and Serpent at different fuel burnup steps and compared with the benchmark data obtained with HELIOS. The comparison was illustrated in Tables 2 and 3 for the transport, absorption and fission production (nu*fission) cross sections at the burnup of 0.15 GWd/tHM. It was found that the cross sections generated by TRITON and Serpent generally compared well within few percent with the reference data obtained with HELIOS. However, the maximum deviations of the TRITON and Serpent results from the benchmark data were within 10-15% for the transport cross sections. Since the same geometry and the same material composition were used in the calculations, it might be mainly due to different calculation methods and nuclear data libraries used in each code. It is also noted that the neutron cross sections obtained with TRITON and Serpent agreed well within ~3% to each other.

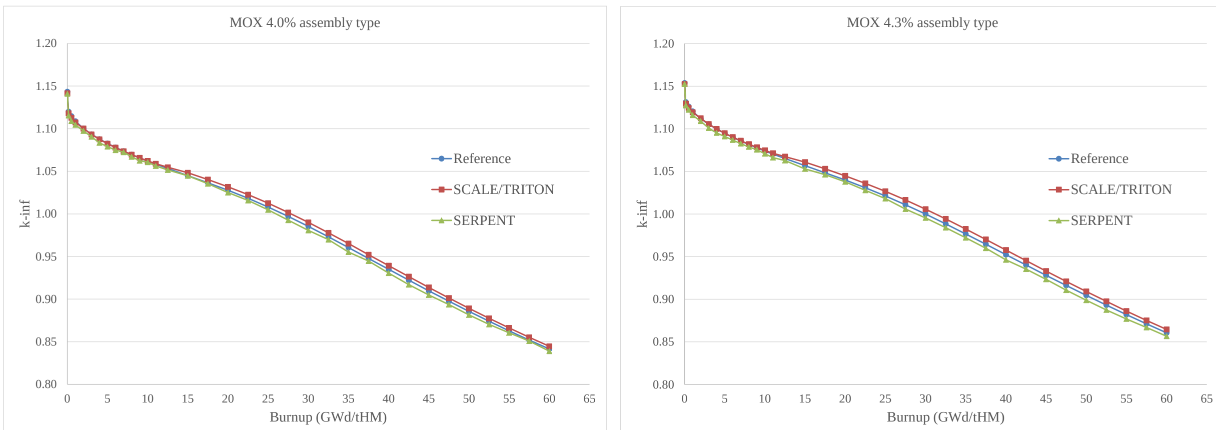

Fig. 2 The k-inf of the MOX 4.0% (left) and MOX 4.3% (right) assemblies versus burnup

Table 2 Two-group cross sections of the MOX 4.0% assembly at the burnup of 0.15 GWd/tHM

	Group	Transport	Absorption	Nu*Fission
Reference	1	2.35776E-01	1.17228E-02	8.30529E-03
SCALE/TRITON	1	2.14159E-01	1.20597E-02	8.16060E-03
SERPENT	1	2.13426E-01	1.19156E-02	8.34383E-03
Reference	2	8.65492E-01	2.50779E-01	3.68848E-01
SCALE/TRITON	2	9.98765E-01	2.41033E-01	3.61891E-01
SERPENT	2	9.94692E-01	2.49328E-01	3.65436E-01

Table 3 Two-group cross sections of the MOX 4.3% assembly at the burnup of 0.15 GWd/tHM

	Group	Transport	Absorption	Nu*Fission
Reference	1	2.35770E-01	1.19080E-02	8.69247E-03
SCALE/TRITON	1	2.14033E-01	1.23033E-02	8.73070E-03
SERPENT	1	2.13148E-01	1.20894E-02	8.71826E-03
Reference	2	8.67568E-01	2.59391E-01	3.84971E-01
SCALE/TRITON	2	1.00603E+00	2.53248E-01	3.71603E-01
SERPENT	2	1.00364E+00	2.57423E-01	3.80709E-01

IV. CONCLUSIONS

In this paper, the SCALE/TRITON (deterministic method) and Serpent (Monte Carlo method) codes were utilized to generate the neutron cross sections for the PWR MOX 4.0% and

4.3% fuel assemblies. The purpose was to use both the deterministic and Monte Carlo methods to prepare the group constants for analyzing REAs in PWR MOX fueled cores using 3D reactor kinetics codes like PARCS and NODAL3. The k-inf values of the MOX assemblies versus burnup and the respective neutron cross sections obtained with TRITON and Serpent were analyzed and compared with the reference benchmark data obtained with the HELIOS code. The comparative results generally show a good agreement between the three codes, demonstrating the reliability of the TRITON and Serpent models developed herein for the PWR MOX assemblies. These calculation models are being improved and applied to transient analyses of PWR MOX fueled cores using 3D reactor kinetics codes.

ACKNOWLEDGMENTS

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TÍNH TOÁN HẰNG SỐ NHÓM CỦA CÁC BÓ NHIÊN LIỆU MOX NẠP TẢI LÒ PHẢN ỨNG PWR DÙNG CÁC CHƯƠNG TRÌNH TÍNH TOÁN SCALE VÀ SERPENT

Vũ Thanh Mai[1]*, Trần Việt Phú[2], Phạm Như Việt Hà[2]

[1]*Đại học Khoa học Tự Nhiên, Đại học Quốc Gia,*
334 Nguyễn Trãi, Thanh Xuân, Hà Nội

[2]*Viện Khoa học và Kỹ thuật Hạt nhân, Viện Năng lượng Nguyên tử Việt Nam,*
179 Hoàng Quốc Việt, Cầu Giấy, Hà Nội

**E-mail:* mai-vu@hus.edu.vn

Tóm tắt: Báo cáo trình bày việc áp dụng các chương trình SCALE/TRITON (dựa trên phương pháp tất định) và Serpent 2 (dựa trên phương pháp Monte Carlo) để tính toán hằng số nhóm của các bó nhiên liệu MOX (nhiên liệu dạng hỗn hợp ô xít) nạp tải lò phản ứng PWR phục vụ các tính toán chuyển tiếp vùng hoạt lò PWR nạp tải nhiên liệu MOX trong các điều kiện vận hành bình thường và sự cố bật thanh điều khiển ra khỏi vùng hoạt. Các bó nhiên liệu MOX nạp tải lò PWR được mô hình hóa với TRITON và Serpent trong đó hệ số nhân neutron vô hạn của các bó MOX phụ thuộc độ sâu cháy của nhiên liệu cùng với các tiết diện neutron được tính toán và so sánh với các giá trị tương ứng của bài toán benchmark. Kết quả so sánh nói chung cho thấy sự phù hợp tốt giữa các giá trị tính toán dùng TRITON và Serpent với các dữ liệu benchmark. Vì vậy các mô hình bó nhiên liệu MOX nạp tải lò PWR được phát triển ở đây dùng TRITON và Serpent có thể được áp dụng để tính toán các hằng số nhóm phục vụ các tính toán chuyển tiếp vùng hoạt lò PWR nạp tải nhiên liệu MOX.

Từ khóa: bó nhiên liệu MOX nạp tải lò PWR, hằng số nhóm, SCALE, Serpent