Overview of ORIGEN-ARP and its Application to VVER and RBMK

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INTRODUCTION

An accurate treatment of neutron transport and depletion in modern fuel assemblies, which are characterized by heterogeneous, complex designs, and higher fuel enrichments and burnups, requires the use of advanced and complex computational tools capable of simulating multidimensional geometries. Most of the modern computational systems developed for this purpose are built through coupling of multidimensional neutron transport and point depletion codes. For many routine applications, the use of these types of tools may not be efficient, as they may require large computational time, significant computer resources, additional specialized knowledge on the methods used by the code or more extensive training of the code user. In these cases, depending on the specific type of application and the level of accuracy required, the user may benefit from the availability of faster and easier to use alternative tools. The ORIGEN-ARP code in SCALE [1] could serve as such an alternative tool for depletion, decay, and source term analyses.

Validation and verification studies of ORIGEN-ARP for various applications have shown that it provides results with an accuracy level comparable to that provided by more complex codes. As in other computational tools, the accuracy level in ORIGEN-ARP is greatly influenced by the adequacy and quality of the data in the cross-section libraries used. The latest release of SCALE includes ORIGEN-ARP cross section libraries generated for a wide range of modern assembly designs that are currently used in the nuclear industry [2]. Results of validation studies for the libraries developed for VVER assembly designs, as well as for more recently generated libraries for RBMK assembly configurations, are presented and discussed in this paper. The validation was performed against isotopic assay measurement data for spent fuel.

DESCRIPTION OF ORIGEN-ARP

ORIGEN-ARP includes a Windows graphical user interface and permits the user to create and execute depletion and decay cases with minimal effort, and to generate tables and plots showing the calculated data in a fast and user-friendly fashion. It has three main components: (1) ARP code to interpolate on a set of pregenerated burnup-dependent cross sections to obtain cross sections for use with ORIGEN-S; (2) ORIGEN-S code to perform depletion and decay simulations; and (3) the OPUS/PlotOPUS codes to extract and plot the calculated results. The burnup-dependent cross-section libraries for VVER and RBMK configurations, as well as many other fuel assembly designs available in SCALE 5.1, were generated using the depletion module TRITON. TRITON performs depletion simulations for two-dimensional (2-D) configurations by coupling the 2-D transport code NEWT with the point depletion and decay code ORIGEN-S. Selected TRITON models for the VVER-440, VVER-1000, and RBMK assemblies are illustrated in Fig. 1. The TRITON calculations employed the 44-group transport library in SCALE based on ENDF-B/V cross section data, and the NITAWL module as a cross-section processor. The libraries generated are representative for the assembly as a whole. More details about the methodology used to develop these libraries can be found elsewhere [2,3,4].

VALIDATION RESULTS

Validation studies of the VVER libraries were performed by using isotopic assay measurement data for spent fuel from experiments carried out at the Khlopin Radium Institute (KRI) in Russia [1]. These measured samples were selected from VVER-440 and VVER-1000 fuel assemblies that were irradiated in reactors operated at the Novo-Voronezh and Kalinin nuclear power plants. The initial fuel enrichments were 3.6 wt % ^{235}U and 4.4 wt % ^{235}U for the samples from VVER-440 and VVER-1000 fuel, respectively. Twenty VVER-440 fuel samples with burnups in the range 20–43 GWd/MTU and 13 samples from VVER-1000 fuel with burnups between 14 and 52 GWd/MTU are considered here. Note that the reported burnup values were determined based on measured values of three burnup monitors: ^{148}Nd, $^{145+146}$Nd, and ^{137}Cs. The decay time, from discharge to the time the measurements were performed, was about 3 to 4 years in the case of the VVER-440 fuel samples and between 7 and 10 years for the VVER-1000 fuel samples. However, the measured nuclide concentrations were back-calculated and reported at time of discharge. The measured data were available for the important actinides and two burnup-indicating nuclides; ^{48}Nd and ^{137}Cs. These two isotopes, as well as ^{237}Np, ^{243}Am, and ^{244}Cm, were not measured in all samples. In particular, ^{137}Cs was not measured in the VVER-440 fuel samples, and ^{237}Np was not measured in any of the VVER-1000 samples.

The performance of the RBMK libraries was assessed by comparison of calculations to isotopic assay experimental data obtained from measurements carried

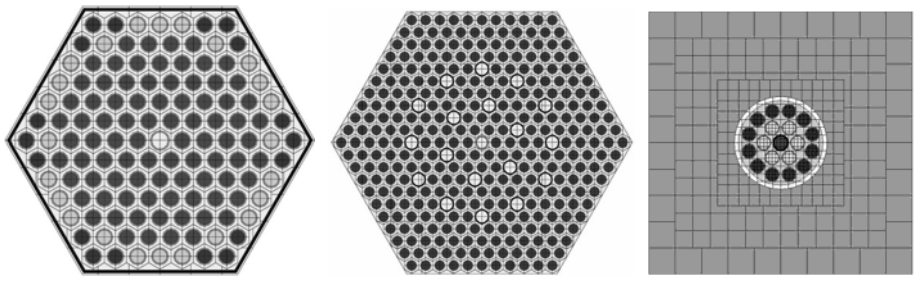
out by KRI on spent fuel samples selected from ten RBMK-1000 assemblies irradiated in reactors operated at the Leningrad nuclear power plant [4]. A set of 15 spent fuel samples were considered for this comparison. The samples had initial enrichments of 1.80, 2.00, 2.02, and 2.09 wt % ^{235}U, and reported burnups in the range 6–23 GWd/MTU. The measurements included uranium, plutonium, curium, americium, cesium, and neodymium isotopes, and the measured concentrations were adjusted by calculation and reported at discharge time.

Simulations with ORIGEN-ARP were carried out for each of the 33 VVER samples and 15 RBMK samples considered. No detailed information on the irradiation history was available, but the total irradiation time, the reported sample burnup, and, in case of the RBMK samples, the axial height of the sample in the fuel rod. The power used in simulations was calculated as a ratio of the reported sample burnup to the total irradiation time. Assumptions were used for the initial concentrations of ^{234}U and ^{236}U in fuel, as these values were not available for most of the samples. The coolant density data for the RBMK samples was not available and was estimated [4].

The results of the calculation-experiment comparison are summarized in Table I, which shows for each nuclide: the number of samples; the average over all measured samples of the calculated-to-experimental concentration ratio (C/E); and the corresponding relative standard deviation (RSD).

The results of the comparison show good agreement between calculations and measurements, especially if one considers that these libraries were generated for typical configurations and are representative of the assembly as a whole. Note that the use of these libraries for simulations of single fuel pins in an environment with physics characteristics different from those of the average assembly (e.g., close to gadolinia-bearing pins), may lead to larger differences between measured and calculated isotope concentrations. In such cases, if improved accuracy is desired, the user may generate cross-section libraries for specific fuel locations in an assembly by following the procedures presented in the SCALE manual.

As shown in Table I and illustrated in Fig. 2, the VVER-440 results for uranium nuclides are good, with ^{235}U overpredicted by 4% on average and ^{236}U calculated within less than 1% of measurement. With the exception of ^{238}Pu, which is underestimated by 5%, the other plutonium nuclides are overpredicted by 3 to 7%. The minor actinides ^{237}Np, ^{243}Am, and ^{244}Cm are overpredicted by 18, 33, and 17% on average, respectively; the variation (RSD) round the average value is similar to the average bias.



VVER-440 VVER-1000 RBMK

Fig. 1. TRITON models for selected VVER and RBMK assembly configurations.

Table I. Comparison of calculated and measured nuclide concentrations

Nuclide	VVER-440			VVER-1000			RBMK-1000		
	# samples	$(C/E)_{avg}$	RSD (%)	# samples	$(C/E)_{avg}$	RSD (%)	# samples	$(C/E)_{avg}$	RSD (%)
^{234}U	20	0.89	4.3	13	0.91	18.1	15	0.96	24.1
^{235}U	20	1.04	5.1	13	0.93	6.5	15	0.98	8.7
^{236}U	20	1.00	1.7	13	0.97	3.2	15	0.90	20.3
^{238}U	20	1.00	0.1	13	1.00	0.2	15	1.00	0.5
^{238}Pu	20	0.95	6.5	13	0.83	14.7	15	0.83	21.4
^{239}Pu	19	1.06	5.3	13	0.95	9.5	15	0.98	8.7
^{240}Pu	20	1.04	3.5	13	0.96	7.6	15	1.07	10.1
^{241}Pu	19	1.03	4.4	13	0.95	13.0	15	1.05	10.6
^{242}Pu	20	1.07	6.5	13	0.99	13.5	15	1.15	21.5
^{237}Np	10	1.18	23.6				15	0.79	32.8
^{243}Am	19	1.33	13.7	7	1.08	24.7	14	1.67	124.1
^{242}Cm							14	0.90	29.9
^{244}Cm	19	1.17	16.7	7	0.93	24.9	14	0.90	30.2
^{137}Cs				10	1.01	2.4	13	1.03	4.0
^{148}Nd	6	0.98	1.5	7	1.05	8.8	15	1.04	3.7

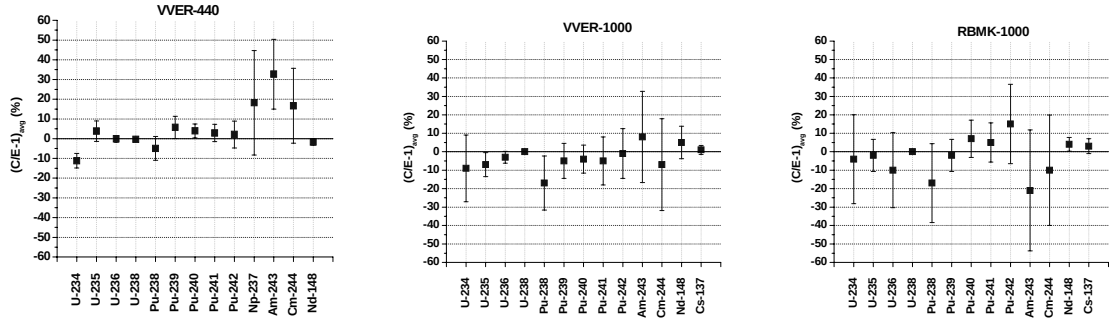


Fig. 2. Comparison of calculated and measured nuclide concentrations.

The burnup-indicator nuclide ^{148}Nd is underestimated by 2%. Note that the effects of the irradiation history details on the calculated concentration were not accounted for because detailed irradiation data were not available. Also, 8 of the 20 VVER-440 fuel samples were selected from fuel pins located at the edge of the assembly (subject to more moderation than the average pin). In the case of some isotopes, it is the contributions of these peripheral pins that cause the C/E ratio of all 20 samples to be overestimated; for example, the C/E for ^{235}U and ^{239}Pu averaged over the 8 peripheral pins is 7 and 10%, respectively. In assessing the performance of the libraries, one must also account for the reported experimental uncertainties, as they can be significant for some isotopes that are difficult to measure. In the cases studied here, the average reported measurement errors (1σ) for the minor actinides ^{237}Np, ^{243}Am, and ^{244}Cm were 7, 6, and 10%, respectively.

The agreement calculation-experiment is also good in the case of the VVER-1000 samples. Major actinides are generally underpredicted, with an average underestimation of 7% and 5% for ^{235}U and ^{239}Pu, respectively. This underestimation can be correlated to the overestimation of the burnup-indicating isotope ^{148}Nd, which is underpredicted by 5% on average, suggesting that the reported burnup for these samples may be too large. An alternative computational approach would be to adjust the reported burnup value used in calculations such that the calculated ^{148}Nd concentration is consistent with the measured value, and in this way the total number of fissions in the calculation is consistent with that corresponding to the measured ^{148}Nd. Unfortunately, ^{148}Nd measurement data was not reported for all samples. The actinides ^{243}Am and ^{244}Cm are on average predicted within 8% of the measured values. Note that for these two isotopes the reported experimental uncertainties are 7 and 4% on average, respectively.

In the case of the RBMK samples, as shown in Table I and illustrated in Fig. 2, the major actinides ^{235}U and ^{239}Pu are predicted on average within 2% of the measurement. The curium nuclides ^{242}Cm and ^{244}Cm are predicted within 10% of the measured data. The difference seen for ^{243}Am is quite large; it is suspected that this discrepancy is largely due to difficulties in the measurements. Note that the average reported experimental errors (1σ) for ^{244}Cm and ^{243}Am are about 10%. Also, the values used for the coolant densities are probably subject to significant uncertainties, as no data were available for the actual RBMK configuration, and estimations were made based on typical BWR axial void profiles [4].

CONCLUSION

ORIGEN-ARP in SCALE 5.1 is a fast and easy to use computational tool for depletion, decay, and source terms analyses, with cross-section libraries available for many fuel assembly designs.

The VVER-440, VVER-1000, and RBMK-1000 cross section libraries for ORIGEN-ARP, generated using the 2-D depletion module TRITON in SCALE, were validated against isotopic assay experimental data for spent fuel. The validation showed good agreement between the measured data and the ORIGEN-ARP results.

REFERENCES

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