VALIDATION OF SCALE DEPLETION MODULE AGAINST VVER EXPERIMENTAL DATA

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ABSTRACT

Some results of validation of the SCALE 4.4 depletion control module SAS2H against experimental data are presented in the paper. The recently published ISTC-DOE report with VVER-440 PIE data for fuel samples from Novovoronezh NPP and ORNL publication with VVER-1000 experimental data for fuel samples from Balakovo NPP have been applied for preliminary validation of depletion calculations. The calculated results for 13 actinides and 14 fission products for VVER-440 samples and for 12 actinides and 5 fission products for VVER-1000 samples have been compared with measured isotopic concentrations.

1. INTRODUCTION

Application of burn-up credit requires knowledge of the reactivity state of the irradiated nuclear fuel. To obtain this knowledge, the isotopic inventory of the irradiated fuel depending on the initial state and irradiation history of the fuel has to be calculated by depletion codes. The depletion analysis results are a necessary input to the criticality safety analysis of the spent fuel management systems with burn-up credit implementation. Three sources of uncertainties have to be taken into account:

- Knowledge of the fuel's initial isotopic state;
- Irradiation history;
- Calculations of spent fuel final isotopic state.

The objective of this work is to validate the SCALE4.4 modular code system [1] against the VVER experimental data for the isotopes given in both the ORNL publication for VVER-1000 [2] and ISTC-DOE report for VVER-440 [3].

The control module SAS2H has been used for calculation of the isotope inventory of VVER spent fuel. SAS2H runs a series of computer codes for depletion analysis. The fuel cross-sections vary with burn-up due to nuclide concentration changing and resulting shift in

the energy spectrum of the neutron flux. The neutron depletion procedure is applied by SAS2H to produce burn-up dependent cross-sections libraries.

2. ASSEMBLY DATA

2.1 VVER-1000 data

Some post-irradiation data are available for the fuel assembly discharged from Unit 3, Balakovo NPP [2]. Two samples of fuel rod 23 (912 and 581), located at different heights, were irradiated during three fuel cycles up to the burn-up level of 47.3 GWd/tU. A geometry model of a VVER-1000 assembly is shown in Fig. 1.

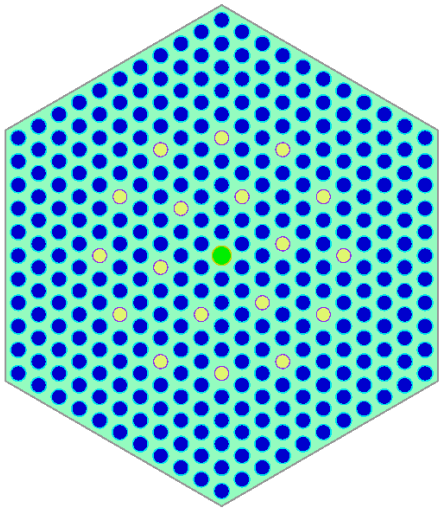

Fig. 1. VVER-1000 fuel assembly

2.2. VVER-440 data

The data for VVER-440 fuel assembly No.13626135 operated during four fuel cycles at the Unit 4, Novovoronezh NPP are taken from Ref. [3]. Fuel sample 57, taken from 2150 mm from the lower part of fuel rod 69 has the maximum fuel burn-up for this assembly 37.75 GWd/tU.

Some data, like down time between fuel cycles, fuel and cladding temperature and coolant pressure and density are not given in the report. They are taken from Ref. [4]. A geometry model of VVER-440 assembly is shown in Fig. 2.

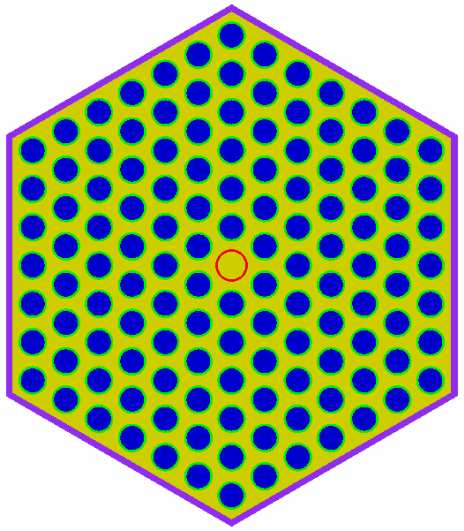

Fig. 2. VVER-440 fuel assembly

3. RESULTS

The isotopic concentrations for the major and minor actinides and fission products with measured concentrations have been calculated.

For VVER-1000 [2] these are 12 actinides (U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, Cm-242, Cm-244) and 5 fission products (Nd-142, Nd-143, Nd-144, Nd-145, Nd-146). For VVER-440 [3] these are 13 actinides (U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243m - isomeric, Am-243, Cm-242, Cm-244) and 14 fission products (Nd-142, Nd-143, Nd-144, Nd-145, Nd-146, Nd-148, Nd-150, Cs-133, Cs-134, Cs-135, Cd-137, Ce-140, Ce-142, Ce-144).

The ratios of calculated to experimental results for various samples are presented in tables. They are also given graphically. The experimental results were reported as kg/t of initial uranium. For comparison purposes, values calculated by SCALE 4.4 have been converted to kg/t of initial uranium.

The comparison of the calculated and experimental results shows that the concentrations of all neodymium nuclides except Nd-144 are well predicted. For VVER-1000 fuel, the Nd-144 concentration is consistently underpredicted whereas for VVER-440 fuel, it is overpredicted. In all cases U-235 is overpredicted. The most significant deviation from the measured values is for Cm-244 for VVER-440 fuel.

The ratios of calculated to experimental results (Calc/Exp) for samples 912 and 581, respectively, from fuel rod 23, irradiated in Unit 3 of Balakovo NPP are shown in Table 1 and Table 2. The boric acid content is 2.10 g/kg. These ratios are also plotted in Figures 3 and 4.

The results for sample 57 of fuel rod 69, irradiated in VVER-440 reactor (Unit 4 of Novovoronezh NPP), at boric acid content 2.61 g/kg are given in Table 3 and plotted in Figure 5.

Table 1. SCALE 4.4 results for VVER-1000 sample 912 at fuel temperature 853 K [kg/t initial Uranium]

Isotope	Calculated	Experimental	Calc / Exp
U-235	11.196	10.180	1.000
U-236	5.643	6.150	0.918
U-238	923.251	925.480	0.998
Pu-238	0.271	0.320	0.847
Pu-239	6.885	6.260	1.100
Pu-240	2.641	2.620	1.008
Pu-241	1.801	1.760	1.023
Pu-242	0.677	0.750	0.903
Am-241	0.057	0.054	1.062
Am-243	0.170	0.151	1.124
Cm-242	0.020	0.021	0.960
Cm-244	0.062	0.059	1.044
Nd-142	0.028	0.034	0.830
Nd-143	1.054	1.0556	0.998
Nd-144	1.305	1.797	0.726
Nd-145	0.876	0.899	0.974
Nd-146	0.939	0.961	0.997

Table 2. SCALE 4.4 results for VVER-1000 sample 581 at fuel temperature 891 K [kg/t initial Uranium]

Isotope	Calculated	Experimental	Calc / Exp
U-235	9.278	8.180	1.134
U-236	5.801	6.340	0.915
U-238	920.993	925.660	0.995
Pu-238	0.316	0.330	0.958
Pu-239	6.569	6.080	1.080
Pu-240	2.777	2.680	1.036
Pu-241	1.856	1.760	1.055
Pu-242	0.826	0.880	0.939
Am-241	0.057	0.071	0.803
Am-243	0.218	0.192	1.135
Cm-242	0.023	0.023	1.000
Cm-244	0.087	0.079	1.101
Nd-142	0.034	0.037	0.919
Nd-143	1.095	1.099	0.996
Nd-144	1.479	1.938	0.763
Nd-145	0.944	0.922	1.024
Nd-146	1.036	1.007	1.029

Table 3. SCALE 4.4 results for VVER-440 sample 57
[kg/t initial Uranium]

Isotope	Calculated	Experimental	Calc / Exp
U-235	11,8153	9,52	1,241107
U-236	4,91181	4,85	1,012744
U-238	1054,94	937,4	1,125391
Pu-238	0,16795	0,164	1,024065
Pu-239	6,8107	6,019	1,131534
Pu-240	2,52342	2,361	1,068793
Pu-241	1,52756	1,489	1,025893
Pu-242	0,51312	0,557	0,921227
Am-241	0,05131	0,0557	0,921227
Am-242m	0,00107	0,00103	1,040602
Am-243	0,10634	0,1032	1,030408
Cm-244	0,03004	0,0486	0,618204
Cm-245	0,00122	0,00135	0,900217
Nd-142	0,02051	0,0206	0,995537
Nd-143	0,93679	0,864	1,084245
Nd-144	1,0887	0,869	1,252819
Nd-145	0,77644	0,777	0,999275
Nd-146	0,78741	0,695	1,132961
Nd-148	0,41776	0,4	1,044392
Nd-150	0,20002	0,187	1,069609
Cs-133	1,29969	1,253	1,037261
Cs-134	0,12828	0,135	0,950229
Cs-135	0,49371	0,449	1,099583
Cs-137	1,4094	1,323	1,065307
Ce-140	1,40096	1,285	1,090243
Cs-142	1,29969	1,178	1,1033
Cs-144	0,38315	0,533	0,718864

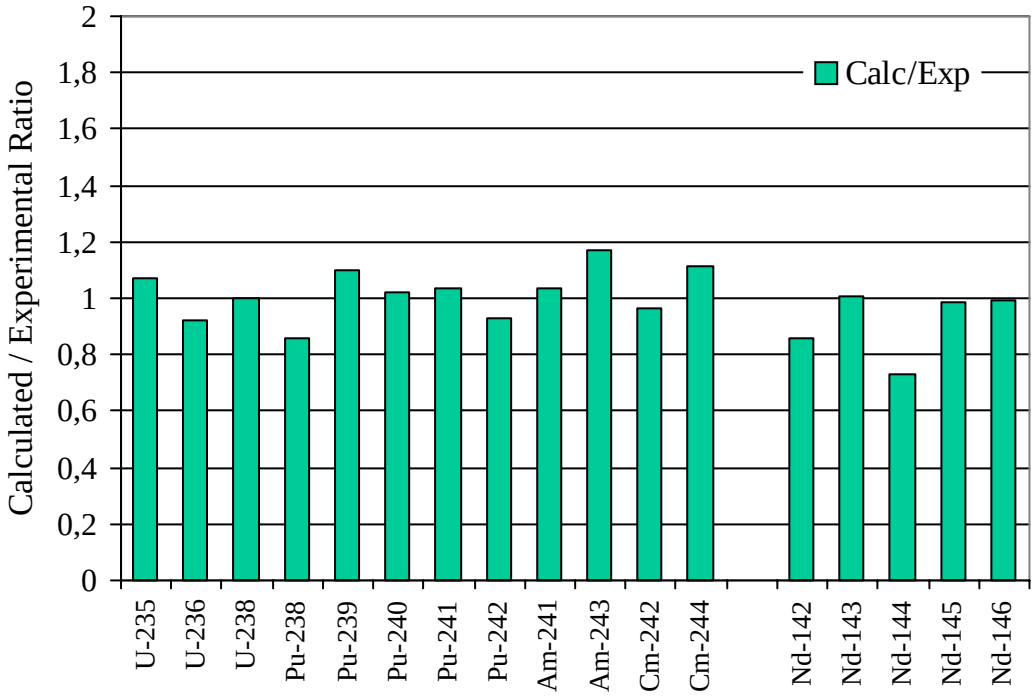

Fig. 3. Calculated / Experimental ratio for VVER-1000 sample 912

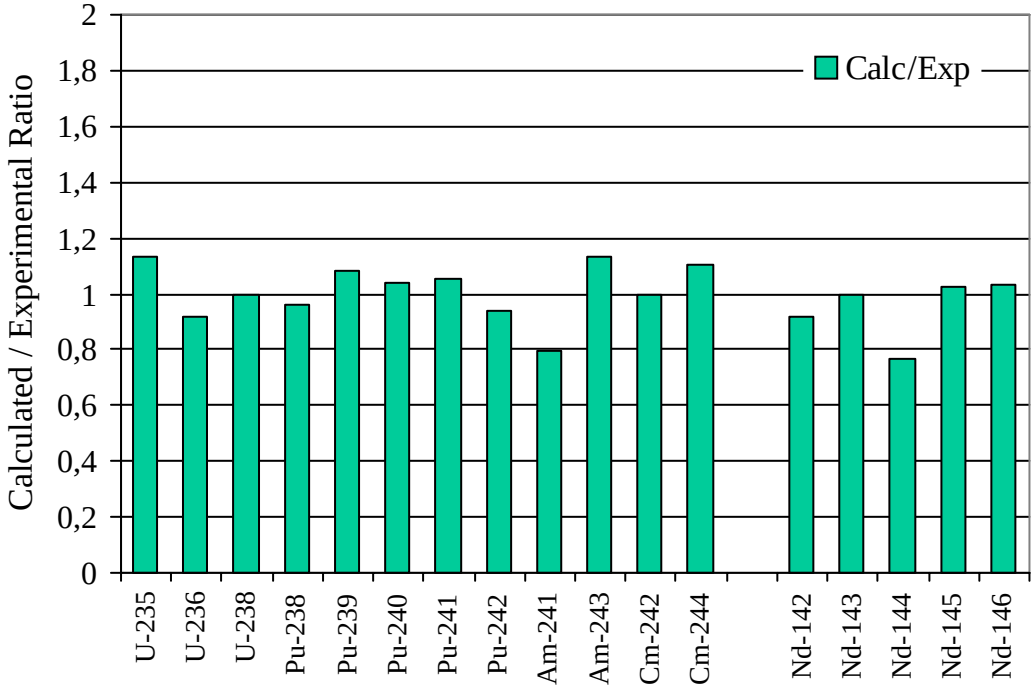

Fig. 4. Calculated / Experimental ratio for VVER-1000 sample 581

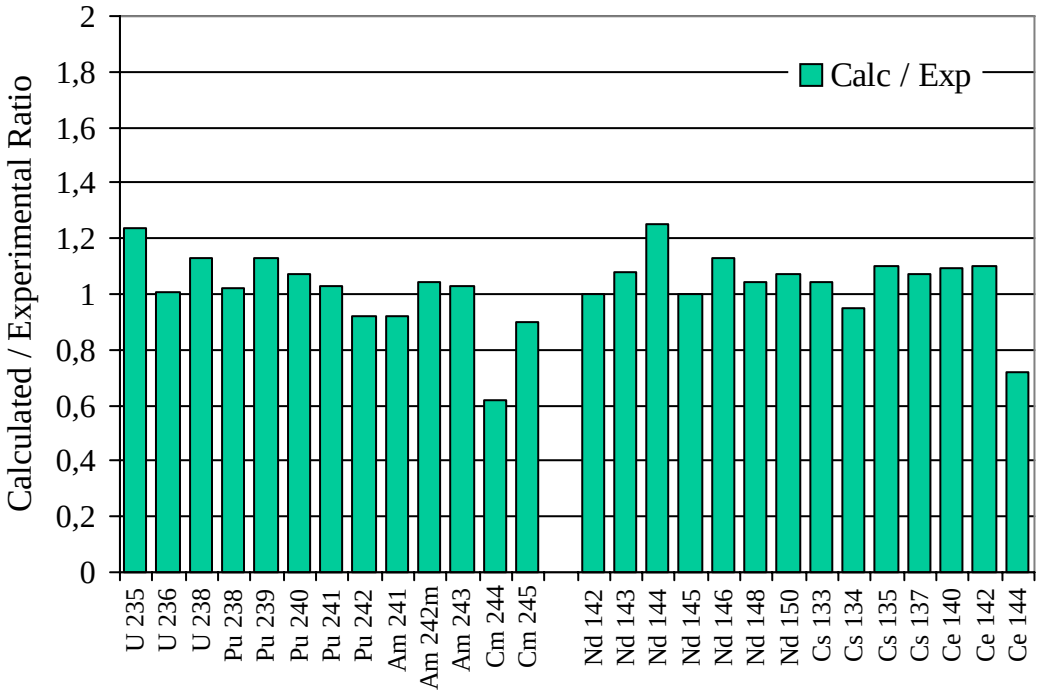

Fig. 5. Calculated / Experimental ratio for VVER-440 sample 57

4. CONCLUSIONS

Some results of validation of the SCALE 4.4 depletion control module SAS2H against experimental data are presented. The recently published ISTC report with VVER-440 PIE data for fuel samples from Novovoronezh NPP and ORNL publication with VVER-1000 experimental data for fuel samples from Balakovo NPP have been applied for preliminary validation of depletion calculations. The calculated results for 13 actinides and 14 fission products for VVER-440 samples and for 12 actinides and 5 fission products for VVER-1000 samples have been compared with measured isotopic concentrations.

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On the basis of the results presented and on the comparison with the experimental data it could be concluded that the SCALE 4.4 system can predict the spent fuel nuclide concentrations necessary for burn-up credit analysis reasonably well.

LIST OF NOMENCLATURE

INRNE	Institute for Nuclear Research and Nuclear Energy
VVER	water-water power reactor
PIE	post irradiation experiment
NPP	nuclear power plant
ORNL	Oak Ridge National Laboratory
ISTC	International Science and Technology Center – Moscow
DOE	Department of Energy USA

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