APPLICATION OF SCALE 4.3 CODE SYSTEM TO THE VVR-S REACTOR SPENT FUEL STORAGE ANALYSIS

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ABSTRACT

The paper presents the methods developed based on the SCALE4.3 code system and calculations carried out to analyse the possibility of safe storage of the Romanian VVR-S reactor spent fuel load inside the CASTOR MTR2 cask. A method has been developed to derive specific problem-dependent nuclear libraries, defined for groups of fuel assemblies. The SAS2H calculation module has been used to simulate different irradiation histories of all spent fuel assemblies and to process the resulting libraries for input to ORIGEN-S code depletion-decay calculations. Using this procedure the spent fuel characteristics of the fuel assemblies for six cooling times have been calculated. The CSAS6 module with KENO-VI criticality code based on the 44-group energy cross-sections library has been used to solve criticality aspects. The sequence SAS4 with MORSE Monte Carlo code has been used for gamma and neutron dose rate estimates. Obtained results are presented and discussed.

Key Words: Spent fuel, Storage, Safety analysis

1 INTRODUCTION

The dry transport/storage container CASTOR MTR 2 [1] is a cast iron cylinder with two stainless-steel lids. The interior is completely occupied by a cylindrical aluminium body with seven cylindrical loading channels, one of them is central and the other six are arranged equally spaced around the central one. Loading units carrying the fuel assemblies will be inserted into each of these loading channels. A loading unit is an aluminium cylinder with one central boron rod and some internal holes tailored in accordance with the geometrical shape of the fuel assemblies. Most of the fuel assemblies of the Romanian VVR-S reactor have a square outer shape, disturbed by 1, 2, or 3 bevelled-corners and the others have a right square shape. According to this different shape of fuel assemblies, there are two types of loading units: Type B carrying four fuel assemblies with a right square shape, and Type C for six fuel assemblies with the disturbed square shape. Independently on the geometrical outer shape, there are two types of fuel assemblies differing by the rod type: The EK10 fuel assembly consists of 16 rods with MgO and UO_2 (10% enriched uranium), and the S36 fuel assembly consists of 15 rods containing a U-Al alloy (36% enriched uranium). For the evaluation of the possibility of storage of the Romanian VVR-S reactor spent fuel in the CASTOR MTR2 cask is necessary to study if the safety criteria are observed.

Calculation parameters necessary to be evaluated and analyzed in order to demonstrate the compliance with the safety requirements under both normal and accident conditions are connected with: i) criticality status and ii) shielding against the radiations. The investigations mentioned above need upstanding, internationally accepted methods, computer codes and nuclear data. From this reason SCALE 4.3 code system that provides several analytical sequences (modules that link codes and data) for use in analysis for the nuclear fuel packages has been implemented and used for solving the problem. To be on the safer side, conservative assumptions have been done. Thus the paper presents especially the results related to the C variant of the CASTOR MTR2 cask

2 SPENT FUEL CHARACTERISTICS

Determination of the isotopic composition of the materials present in the VVR-S reactor spent fuel and subsequently derivation of the heat generation and radiation source terms has been the first study performed. Further analyses of the chosen storage option such the evaluation of the burn-up credit, the shielding and the heat transfer studies have to be based on the derived amounts mentioned above. Additionally, detailed characterization of the entire radioactive inventory as a function of time is required for the knowledge of the time evolution of the fuel status during the storage.

2.1 Derivation of the VVR-S reactor problem dependent libraries

SAS2H module [2] of the SCALE 4.3 system has been applied to produce several time dependent libraries as a function of the specific VVR-S reactor design characteristics, operating parameters and material composition for input to ORIGEN-S [3] cases.

The procedure developed to model the different distributions of irradiation histories of all spent fuel assemblies of a specific type has the following steps:

setting-up of the groups of fuel assemblies that have common irradiation history;

processing of the specific burn-up dependent library (SAS2H) for each group, by using data of a representative fuel assembly inside the group;

performing the depletion-decay calculations (ORIGEN-S) for all fuel assemblies component of the group by using as input the corresponding dependent library.

The groups with common irradiation history have been established based on the fuel irradiation analysis. Averaged power has been considered the relevant parameter that identifies the complicated irradiation history of each fuel assembly (see Fig.1 and Fig. 2)

The representative fuel assembly of each group used for the simulation with SAS2H has been defined by an averaged power calculated as the arithmetic mean of the real values of the fuel assemblies inside the group (see the histograms of the Fig. 3 and Fig. 4) and an irradiation time given by the maximum operation time of the group components.

According to the code requirement the entire residence time at the representative power has been splitted in cycles of 1000 days. Based on this method, for EK-10 fuel assemblies the associated dependent libraries have been produced for four groups:

group I containing 5 fuel assemblies with averaged power <0.01 MW/assembly;

group II containing 33 fuel assemblies with averaged power $\in(0.01, 0.015)$ MW/assembly;

group III containing 29 fuel assemblies with averaged power $\in(0.015, 0.02)$ MW/assembly;

group IV containing 6 fuel assemblies with averaged power >0.02 MW/assembly;

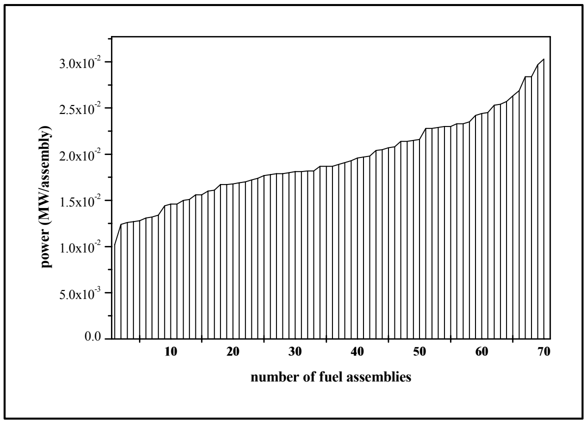


Figure 1. Averaged power distribution of the EK-10 fuel assemblies

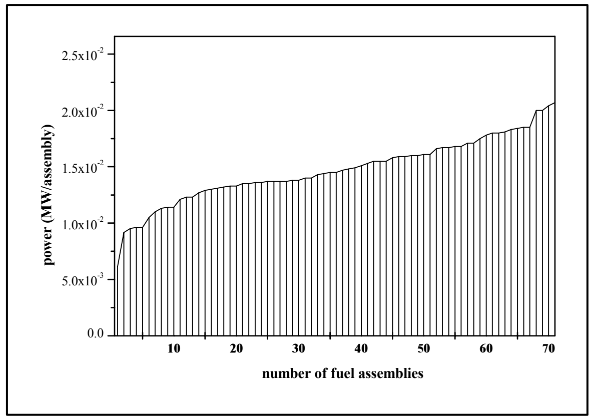


Figure 2. Averaged power distribution of the S-36 fuel assemblies

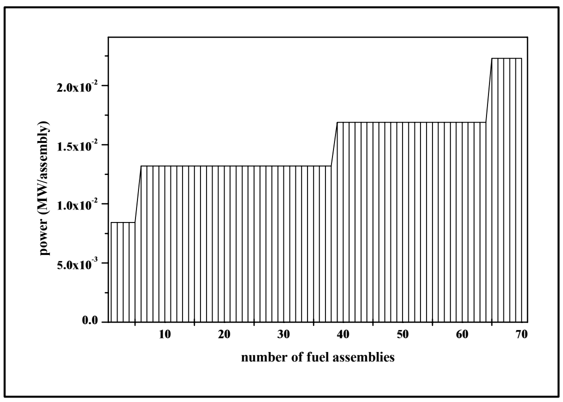


Figure 3. Averaged power distribution of the EK-10 fuel representative assemblies

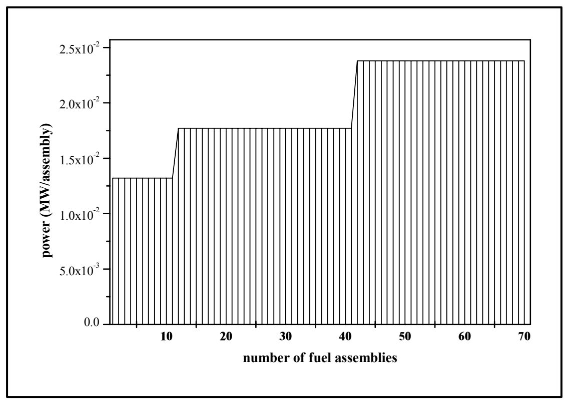


Figure 4. Averaged power distribution of the S-36 fuel representative assemblies

Derived oriented libraries for the specific S-36 VVR-S spent fuel assemblies have been obtained for other three groups defined as follows:

group I containing 12 fuel assemblies with averaged power <0.015 MW/assembly;

group II containing 30 fuel assemblies with averaged power $\in$(0.015, 0.02) MW/assembly;

group III containing 28 fuel assemblies with averaged power >0.02 MW/assembly.

2.2 Calculation undertaken

Using the created dependent libraries ORIGEN-S calculations have been performed to compute radionuclide inventory, thermal powers, gamma source spectra and neutron source strength. The assessment of an averaged burn-up for 79 EK-10 fuel assemblies irradiated during the first period (7-8 years) of the reactor operation has been necessary because not primary data (irradiation history, burn-up, etc.) have been available. Based on this, an ORIGEN-S calculation has been performed using a dummy fuel assembly representing an averaged behaviour (residence in reactor and cooling time).

Calculations have been done for the reference time moment of the reactor shut down and supplementary five reference decay time of 5, 10, 25, 50 and 100 years after the accounted reference time moment.

2.3 Results and Discussions

Resulted inventory of the total activities calculated at the reference shut down are shown in Fig. 5, for EK-10 spent fuel assemblies and respectively Fig. 6 for S-36 fuel type.

As have been expected the study of the radionuclide inventory has been shown that the major contribution to the fuel assembly activity is due to the fission products. The activity of the actinides has low levels, while the contribution of the light materials to the total activity is negligible. Low maximum thermal power value of 2W for EK-10 fuel assemblies and respectively of 3.98W for the S-36 fuel type have been obtained at the reference time moment. From the analyse of the thermal power results the conclusion that the heating transfer analysis is not necessary to be performed has been drown.

The detailed quantities of the spent fuel fissile materials have been also derived for safeguards requirements. Ranges of fuel enrichment values between (3.6 - 8.4)% for EK-10 fuel type and respectively between (18 - 36)% for S-36 fuel type have been found at 100 years cooling time. Gamma and neutron spectra and total strengths have been analyzed and used as source terms for further shielding calculations.

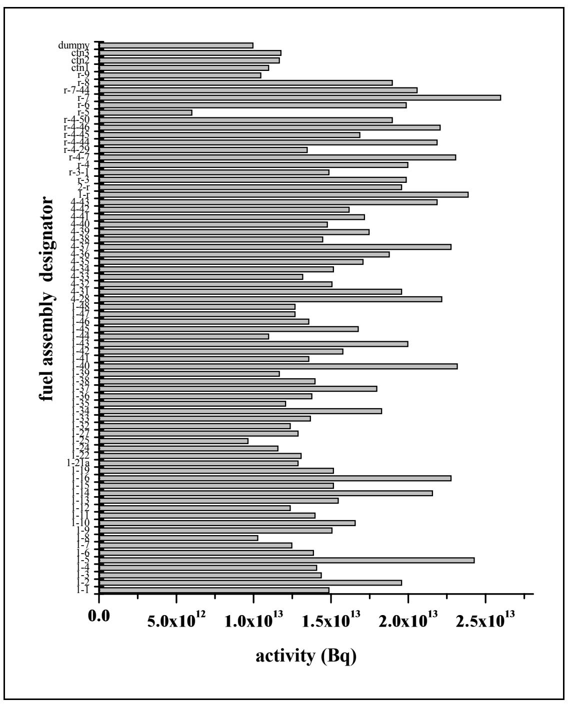


Figure 5. Radioactive inventory of the EK-10 fuel assemblies calculated for the reference time moment

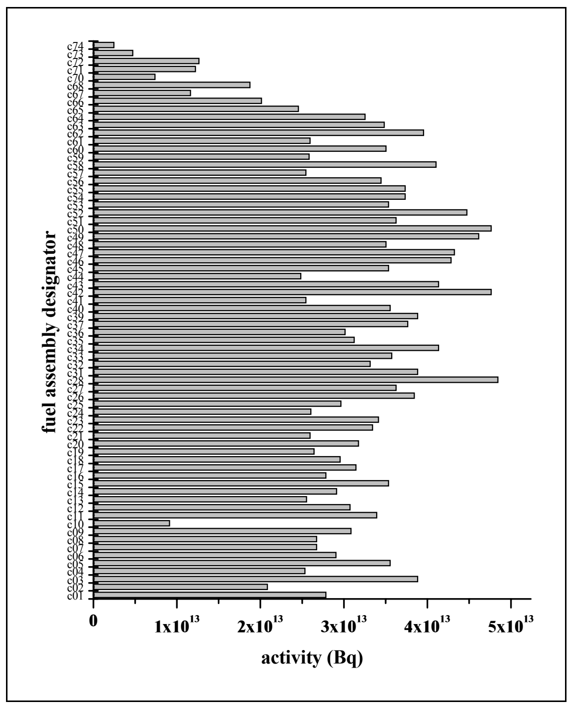


Figure 6. Radioactive inventory of the S-36 fuel assemblies calculated for the reference time moment

3 CRITICALITY SAFETY ANALYSIS

3.1 Modeling. Calculations. Results

Module CSASVI [4] based on KENO-VI Monte Carlo code has been used to study the criticality status of the spent fuel inside the cask.

In order to assure that the cask is sufficiently far from the criticality even for the worst case the calculation model has been assumed: i) the loading of the fresh fuel inside the cask; ii) the use of the reflective boundary condition to simulate an infinite lattice of containers; iii) water flooding of the volumes that are void under normal conditions.

Criticality calculations have been carried out for both types of fuel assemblies based on SCALE 44-group energy cross-section library. A supplementary calculation has been performed by means 238 –group energy library only for the reference case of S-36 fuel type. The obtained results are presented in the Table I.

Table I. Resulted k_{eff} values

Fuel type	**44GROUPNDF5 SCALE4.3 library**	**238GROUPNDF5 SCALE4.3 library**
S-36	0.8901 +/- 0.0019	0.8879 +/- 0.0022
EK-10	0.8753 +/- 0.0019	

4 SHIELDING INVESTIGATIONS

The sequence SAS4 [5] with MORSE, Monte Carlo code, based on the (27n-18g)-coupled group energy cross-sections has been used to study the shielding aspects of the problem.

4.1 Gamma and neutron source calculation

One of the main findings of the radioactive inventory calculations has been the derivation of the gamma and neutron sources. Analysis of the those results has allowed for further discussion in chosen the appropriate values of the radiation source terms needed for the shielding investigations of the cask.

Because, up to now the details of the different CASTOR loading are not yet established it has been considered that the entire loading of a cask consists of fuel assemblies having identical properties with a representative one. Such representative fuel assembly used in calculations has been defined as follows:

- for normal situation, when upper limits of the expected dose rates are desired to be calculated, the dummy assembly is characterized by the maximum values of gamma and neutron sources;

- for loading and audit conditions, when the detailed loading scheme have to be taken into account the conservative criterion is not valuable and therefore an averaged value of both gamma and neutron sources has been taken into account.

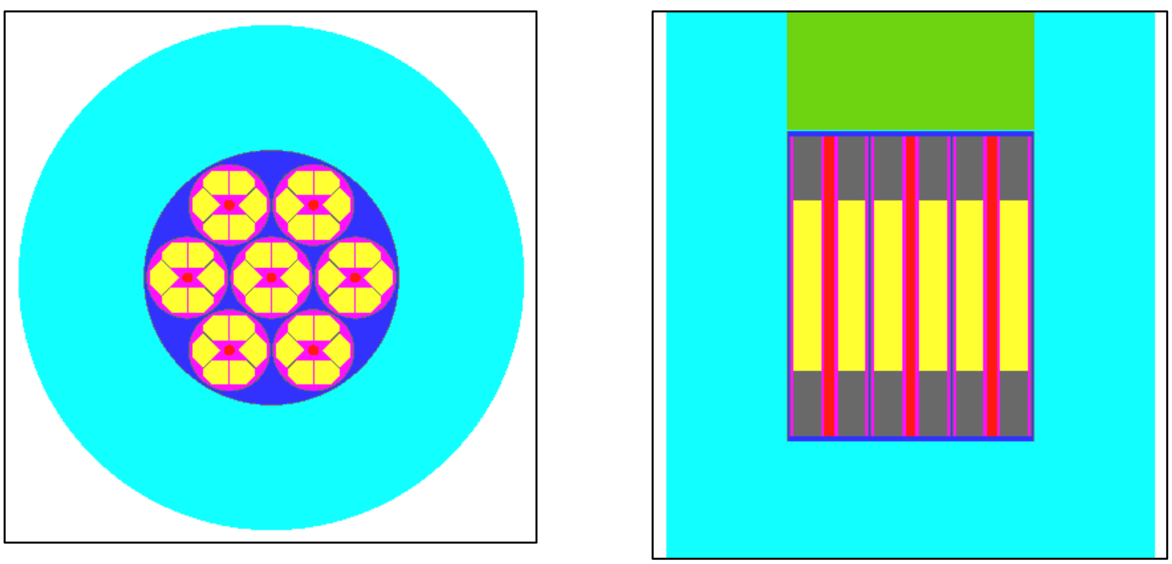
Constant radial and axial distributions for active part of the sources have been assumed.

The contribution of the end parts of the fuel assembly has been considered also for axial dose rate estimates over open cask. With a view to obtaining the gamma source arising from the end of fuel assembly an ORIGEN-S calculation has been performed separately considering only 5g of light structural assemblies.

4.2 Modeling and Calculations

The very complex and non-standard geometry of the cask has not allowed the elaboration of the calculation model with the help of standard casks available in the SAS4 sequence of SCALE4.3 system and therefore the detailed geometry has been built by using MARS [6] module.

Using this utility a very complicated geometry model has been created as input for Monte Carlo MORSE code [7]. The calculation model used assumes (see Fig. 7) that the materials of fuel assemblies and their sources have been smeared over the horizontal cross section of the fuel assembly.



transversal cross section for z=0 **vertical cross section for x, y =0**

Figure 7. CASTOR MTR2. Geometry model for the shielding calculations

Additionally, other approximations of the calculation model done; causing the overestimating of the dose rates can be characterized shortly:

- the wall of the fuel assembly has been added to the surrounding loading unit,
- hence the smearing of the fuel materials has been adapted for the remaining surface;
- symmetric homogeny cylinders with 5 cm height have been modelled of the ends of
- the assembly and the corresponding remained space in the end zones has been filled with air;

- vertical symmetry of the cask has been assumed, replacing the existing slight asymmetry of the cask.

A full characterization of the spent fuel cask has been performed using this calculation model:

- neutron and gamma doses have been computed in separated calculations based on the SAS4 automated biasing procedure;
- different calculations have been performed for radial and axial detectors.

For both gamma and neutron dose rate estimates over the open cask only axial calculations have been performed. A supplementary calculation, for this last configuration has been needed for the estimation of gamma dose contributed by the end part material.

The surface detectors are located radial or axially on the outermost surface of the cask and 1, 2 and 3 m from this outermost surface.

The axial detectors are circular discs, while the radial detectors are side surfaces of cylinders having appropriate radii and heights given by the height of active part of the fuel assembly.

The ANSI standard flux-to dose conversion factors have been used. These response factors correspond to the coupled library (27 neutrons-18 gamma) energy groups used for all the calculations performed.

4.3 Results and Discussions

The obtained results for both closed and open cask configurations are presented in Tables II and III.

Table II. Maximum dose rate values outside the CASTOR MTR 2 cask

Total sources		Dose rates Surface μSv/h FSD*	+1m μSv/h FSD*	+2m μSv/h FSD*	+3m μSv/h FSD*
[γ s^{-1}] 9.7142+14	Radial	68.63 0.02	6.32 0.04	3.22 0.02	1.89 0.02
	Axial	23.32 0.04	4.03 0.04	1.85 0.05	0.97 0.05
[n s^{-1}] 7.8246+4	Radial	2.45-1 0.01	2.19-2 0.01	9.59-3 0.01	5.21-3 0.01
	Axial	2.73-1 0.01	2.51-2 0.01	8.78-3 0.01	4.33-3 0.01

*FSD fractional standard deviation

Table III. Dose rate values over the open CASTOR MTR 2 cask

Total sources	Dose rates Surface Sv/h FSD*	+1m Sv/h FSD*	+2m Sv/h FSD*	+3m Sv/h FSD*
[γ s^{-1}] active part 5.8997+14	1.88 0.05	1.22 0.02	0.64 0.04	0.35 0.05
end parts 1.5107+9	2.29-5 0.02	1.28-5 0.01	6.37-6 0.02	3.73-7 0.02
[n s^{-1}] 3.1391+4	3.66-7 0.01	9.74-8 0.04	4.08-8 0.01	2.20-8 0.02

*FSD fractional standard deviation

The statistical errors (under 5%) are comparatively insignificant to the more important errors that come from modelling limitations. The error due to the smearing of the fuel materials is about 30% [8]. The approximations done for radioactivity and further gamma and neutron sources ORIGEN-S calculation induce an error of about +/-15% per fuel assembly. But, the large number of fuel assemblies of the cask loading (42) allows the compensation of these errors arising from uncertainties given by the various positions of the fuel assemblies during irradiation.

The values of the dose rates exterior to the cask charged with VVR-S spent fuel (see Table II) are below the limit of 2mSv h^{-1}, accepted by the National Regulations for Nuclear Safety.

In comparison with the gamma dose rates arising from active part of the fuel assemblies, those that came from the materials of the end part are negligible (see Table III).

5 CONCLUSIONS

The main achievements of this work are:

A methodology has been developed to obtain and apply problem depended libraries specific to the VVR-S reactor;

Detailed evaluation of the radioactive inventory, thermal power, sources terms has been done based on these oriented nuclear data sets;

Calculation model for criticality safety analysis has been established and calculations have been carried out based on this model;

Calculation model for shielding safety analysis has been established and dose rate evaluation for normal and loading conditions have been carried.

It is concluded from these computational studies that:

Related to the criticality safety, even from the most restrictive conditions the CASTOR MTR 2 cask assures for the VVR-S reactor spent fuel the compliance with the safety condition (K_{eff}<0.95);

From the shielding safety point of view also the CASTOR MTR 2 cask offers a safe containment of the Romanian VVR-S reactor spent fuel. Resulted dose rates exterior to the cask are less then the accepted limit criterion even for the most conservative conditions.

Analyzing the relevant phenomena related to the safety these studies have been demonstrate that the CASTOR MTR2 is a safe option for the VVR-S spent fuel storage.

Additionally the resulted dose rate values will be used to assist the achievement of the working procedures that have to be observed during the loading of the cask as well as for the audit activities. The used conservative approximations have to be taken into account in order to have more precise basic input needed to established the conditions of personnel operation in the working area and to assure the safety of the staff upon the ALARA principles.

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