Spent Fuel Characterization to Support NPP Krško Spent Fuel Dry Storage Project

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ABSTRACT

Activities performed at the Jožef Stefan Institute (JSI) to support NPP Krško Spent Fuel Dry Storage project are presented in this paper. In the initial phase, NPP Krško FAR and JSI FASLIB fuel assembly databases have been harmonized. Validated data from the FASLIB database served to analyse fuel manufacturing and irradiation data. The range of the fuel parameters needed to cover calculation of the fuel irradiation was determined. The model for the TRITON/NEWT calculation module from the SCALE package has been validated by comparison of the predictions obtained by the stochastic neutron transport code Serpent2. The focus was on the NPP Krško spent fuel decay heat, activity and neutral particle emissions for cooling times up to 50 years. Results obtained by both codes are in good agreement. ORIGAMI model running on the TRITON/NEWT binary one-group reaction coefficient libraries (.f33 files) has been validated. The importance of the energy produced by the neutron capture has been stressed to reduce observed differences. NPP Krško spent fuel sensitivity study has been accomplished based on the developed TRITON/NEWT model and fuel predetermined parameters range. It seems that fuel burnup and enrichment are the most influential parameters in the calculation of observables. In the evaluation of the averaging process, non-linearity of the sensitivity coefficients was examined. The effect is mainly important for the enrichment and burnup, where the decay heat prediction might be several percent too low. A few cases based on the real data have been analysed to demonstrate reasonable axial discretization. Planned activities are gradually coming to successful conclusion contributing to the safe and economical spent fuel dry storage on the NPP Krško site.

Keywords: *Spent nuclear fuel, decay heat, activity, photon source term, neutron source term, sensitivity analysis, spent fuel dry storage*

1 INTRODUCTION

Currently, all spent fuel in the NPP Krško (NEK) is stored in the spent fuel pool located in the fuel handling building of the plant. After the Fukushima accident in 2011, it was decided to relocate some of the fuel in the dry storage casks to increase safety and storage capability for spent fuel assemblies. Dry storage concept is safer and more reliable storage solution, compared to spent fuel pool storage concepts, since it uses passive design systems. Within NPP Krško Spent Fuel Dry Storage (SFDS) project activities, the construction of a dry storage building started in March 2021 after a construction permit was issued. The building will host Holtec International HI-STORM FW

casks consisting of a steel canister inserted in a concrete overpack. The relocation of fuel assemblies from the pool to the new storage facility will start with the relocation of about 600 fuel assemblies in 16 casks. The second phase, scheduled to begin in 2028, will fill another 16 casks. The final phase will be completed five years after the power plant is scheduled to shut down in 2048. The storage facility will be built to last 100 years, after which permanent disposal solution will be needed. Jožef Stefan Institute (JSI) is supporting NPP Krško SFDS project together with the Faculty of Electrical Engineering and Computing (FER), University of Zagreb.

The main function of the storage casks is to provide sufficient spent fuel cooling and shielding against gamma and neutron radiation emitted by the radioactive contents. The adequate characterization of the spent fuel, where the decay heat, fuel activity, photon and neutron source term are determined, is an important parameter as it affects the loading of the casks. The loading constraints arise from the safety and design requirements for the specific storage casks. This paper reviews the activities at the JSI to support NEK by providing independent spent fuel characterization data, which will be used to optimize cask loading plan for all spent nuclear fuel generated until the end of NEK commercial operation and optimization of operational safety margins of the spent fuel pool.

In the first phase of the project, fuel assembly database was verified by comparison and harmonization of the plant data with the independent JSI database. These parameters, such as fuel enrichment, burnup, fuel operational conditions during irradiation, number of BPR and IFBA rods, dictates spent fuel behaviour predictions and are needed in the process of the spent fuel characterization. Validation of the calculation models is presented in Section 3. Spent fuel sensitivity study has been accomplished based on the developed TRITON/NEWT model and spent fuel predetermined parameters range. Results and conclusions are given in Section 4. Final segmentation of the fuel taking into account proposed homogenization is demonstrated in Section 5. Please note that mostly decay heat and activities will be presented in this paper due to limited space available, although the photon and neutron emission rates are examined also in the project.

2 FUEL MANUFACTURING AND IRRADIATION DATA

NEK is managing fuel manufacturing and irradiation data with the FAR code [1], while JSI uses CORD-2 system [2]. Both databases were compared to independently verify fuel assembly (FA) data. NEK is regularly updating its database according to the “as built” fuel manufacturing data (provided by the fuel manufacturer Westinghouse Electric Company) and measurements performed at the plant. The FASLIB database maintained at JSI is produced by the CORD-2 package. It should be noted, that the fuel manufacturing data used in calculations are also based on the “as built” data periodically sent from NEK to JSI.

The following FA parameters were compared:

- mass,
- enrichment,
- number of BPRs,
- number of IFBA rods,
- burnup.

All FAs inserted into the core up to the cycle 31 were considered. Burnup comparison was performed for the first 30 cycles. Comparison has shown a few minor differences, which were resolved by inspection of original documents from NEK archive. After harmonization both databases have the same fuel manufacturing data. Final FAs burnup reported in the FAR (obtained from INCORE measurements) and FASLIB (obtained from CORD-2 calculations) agree well. All FAs burnup values are within 3.5 %.

2.1 Determination of the fuel parameter space

Determination of the suitable fuel parameters range covering entire spent fuel variations is needed to define valid parameter space. FASLIB database was used, since it contains calculated data required for the sensitivity analysis such as burnup, various temperatures, moderator density, specific power, soluble boron concentrations etc. Data are listed for 10 axial layers with relative thicknesses (1, 1, 2, 4, 4, 4, 4, 2, 1, 1), where the unit is 15.24 cm (6 inch). Special attention is needed for the first and last axial periphery region, where masses and enrichments differ significantly (axial blankets) compared to the central regions for the Vantage5 fuel.

Relative large parameter variations occurred during the plant operation of almost 40 years. Burnup weighted average boron concentration, FAs were experiencing during the irradiation, is shown in Figure 1. Operating cycle on the x-axes is the first cycle in which particular FA was inserted into reactor core. Each dot on the plot represents one FA. Please note that the same graphical representation is used also in subsequent figures in this section. During the plant lifetime average boron concentration increased due to the longer cycle length - from approximately 500 ppm in initial 12-month cycles to even more than 1100 ppm in last 18-month cycles.

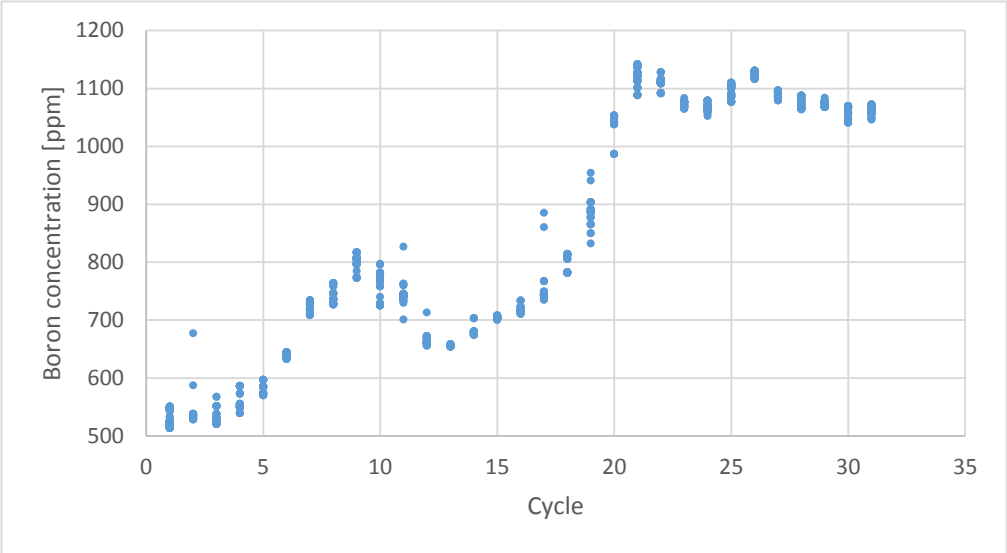

Figure 1: Average soluble boron concentration

Total FA burnup values are presented in Figure 2. Maximal values are around 54500 MWd/tU. During the spent fuel characterization, some averaging of data is performed often. It is therefore interesting to investigate the maximal differences of burnup in the axial regions. Minimal and maximal values for the eight central regions are shown in Figure 3. Overall, maximal value is almost 60000 MWd/tU, while the differences in central regions can be up to ~23000 MWd/tU.

Variations in enrichment are also large. Maximal nominal enrichment is almost 5 %, while the enrichment in some axial blankets is enrichment of natural uranium e.g. ~0.72 %. Based on the evaluation of all data, suitable fuel parameters ranges, that encompass all particular FA values, can be determined (Table 1).

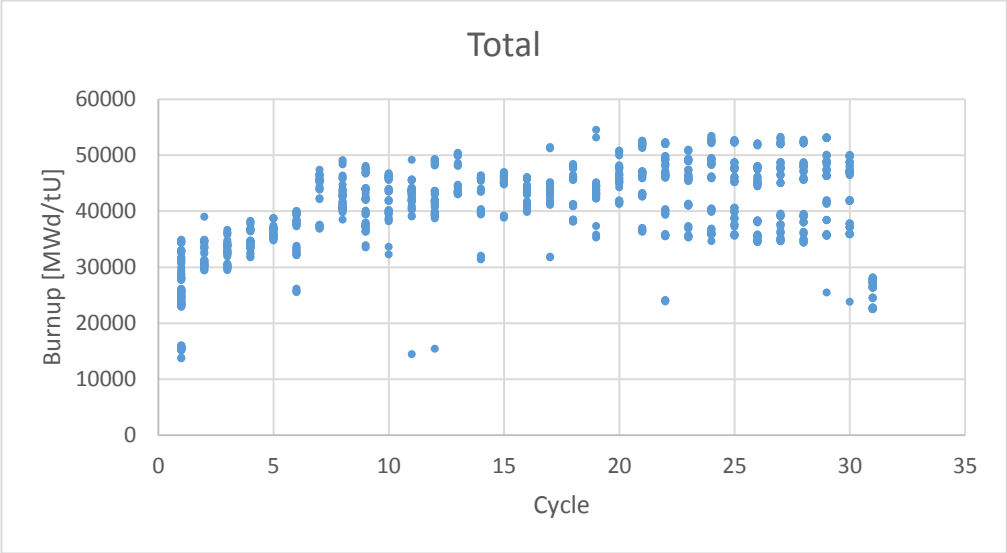

Figure 2: Total fuel assembly burnup

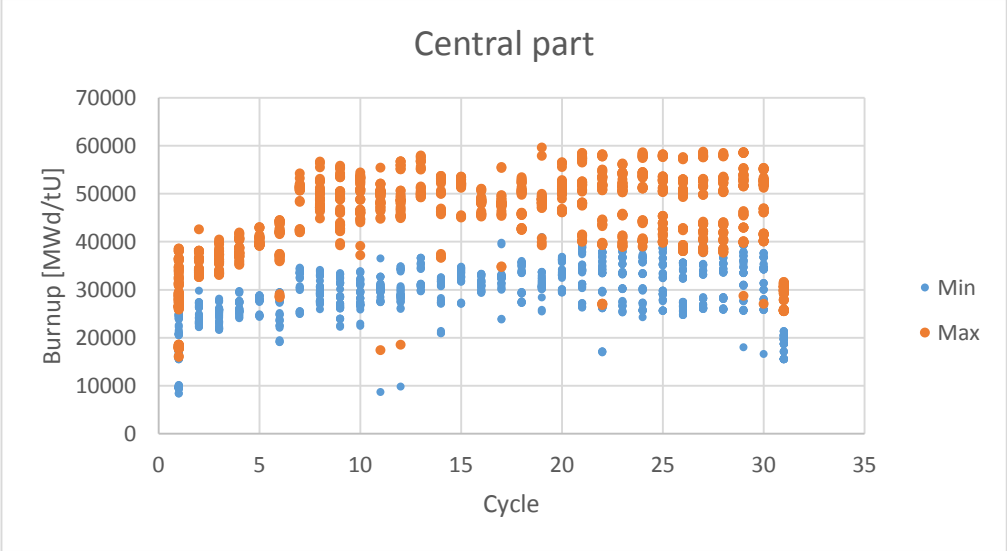

Figure 3: Minimal and maximal burnup values of central regions

Table 1: Fuel parameters range

Parameter	Range
Fuel enrichment	0.7 % - 5 %
Burnup range	36000 MWd/tU - 60000 MWd/tU
Fuel effective temperature	600 K – 1100 K
Moderator density	0.655 g/cm^3 (601.91 K)-0.755 g/cm^3 (558.56 K)
Boron concentration	500 ppm -1200 ppm
Specific power	15 W/gU - 65 W/gU
Number of rods in BPR absorbers	0 - 16
Number of IFBA rods	0 - 116

3 DETERMINATION OF THE SUITABLE CALCULATION TOOLS AND MODELS

SCALE code system [3] has been selected for the spent fuel characterization, while the Serpent2 [4] code served as an independent supplementary tool to validate prepared SCALE models. Both code systems are widely used and are thoroughly validated.

The TRITON/NEWT module from the SCALE package has been used in the analysis, since for now more contemporary POLARIS is not capable to provide ORIGEN binary one-group reaction coefficients library files (.f33). SCALE version 6.2.4 with internal 56 energy group library v7-56 based on the ENDF/B-VII.1 [5] evaluated nuclear data files has been applied. Version 2.1.29 of the Monte Carlo code Serpent2 and the continuous energy cross section library in ACE format, based also on the ENDF/B-VII.1 evaluated nuclear data files [5] has been used for the validation.

3.1 NPP Krško fuel model validation

A typical FA with 4.95 % enrichment and no IFBA rods was selected as a test case. A reference 2-D case scenario (infinite in axial direction) with periodic boundary conditions in lateral direction consists of the following reactor operational parameters:

1. Fuel temperature 900 K,
2. Moderator temperature 580.46 K with density 0.70871 g/cm3,
3. Soluble boron concentration of 1000 ppm.

Parameters are close to the average operational parameters applied in the last NPP Krško cycles. Decay heat, fuel activity, photon and neutron source term after burnup of 60000 MWD/MTU were examined.

Differences in the decay heat (DH) are given in Figure 4. Label Triton denotes relative SCALE differences to the Serpent, while B1 denotes the effect of the B1 approximation (difference between SCALE runs with and without B1 approximation). With the B1 option, depletion of the FA in the critical neutron spectrum obtained by the critical buckling can be performed. TRITON differences are less than 1.3 % for cooling times larger than 1 year. Results agree well with values reported in [6] and [7]. The effect of the B1 approximation is less than 0.1 %. Differences in the activity (A) are given in Figure 5. TRITON differences are less than 1 % for cooling times larger than 1 year. The effect of the B1 approximation is less than 0.05 %. Differences in neutron and photon emission rates are inside 5 % and 2 % respectively, while the effect of B1 approximation is less than 0.3 % in neutron emission and less than 0.2 % in photon emission.

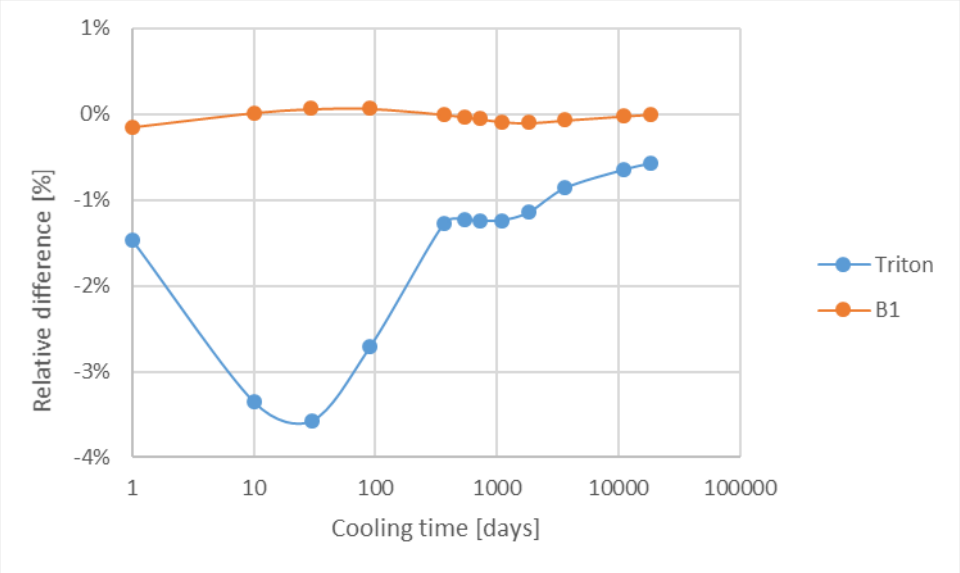

Figure 4: Relative differences of the decay heat – DH; Triton:$(DH_{SCALE} - DH_{Serpent})/ DH_{Serpent}$, B1: $(DH_{B1} - DH_{SCALE})/DH_{SCALE}$

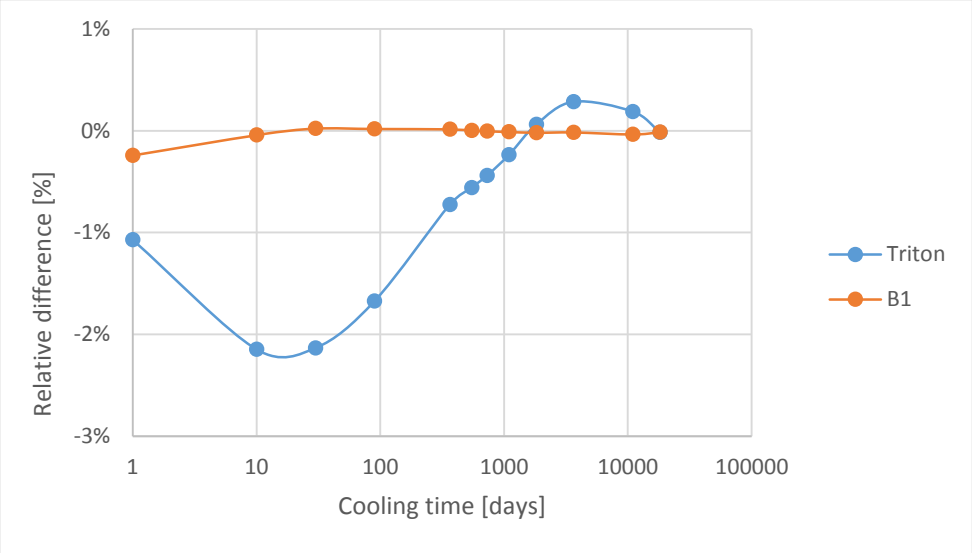

Figure 5: Relative differences of the activity – A; Triton:($A_{SCALE} - A_{Serpent}$)/ $A_{Serpent}$, B1: ($A_{B1} - A_{SCALE}$)/A_{SCALE}

3.2 Origami model validation

The use of the TRITON/NEWT sequence for the characterization of the more than 1400 considered FAs would be impractical, since a single depletion run takes around 2 hours on the PC with the Intel I7 processor. Fortunately, the TRITON/NEWT module is capable to provide ORIGEN binary burnup dependant one-group reaction coefficients library .f33 file for each depletion run. These library files are sufficient to independently determine isotopic composition during the fuel irradiation (solution of Bateman equations) avoiding the need of time-consuming neutron transport calculation. However, the calculation sequence has to be validated.

Results of the TRITON/NEWT calculation have been compared to the ORIGAMI results simulating the same irradiation history and using .f33 files generated by the reference TRITON/NEWT run. It should be noted that the entire ORIGAMI depletion with more than 20 individual ORIGEN irradiation steps takes only around 10 seconds. Relative differences in the decay heat for the FA enrichments from 0.7 % to 5 % are presented in Figure 6 and in the activity in Figure 7. Differences in the decay heat are up to 1.3 %, while the differences in the activity are little smaller (up to 0.7 %).

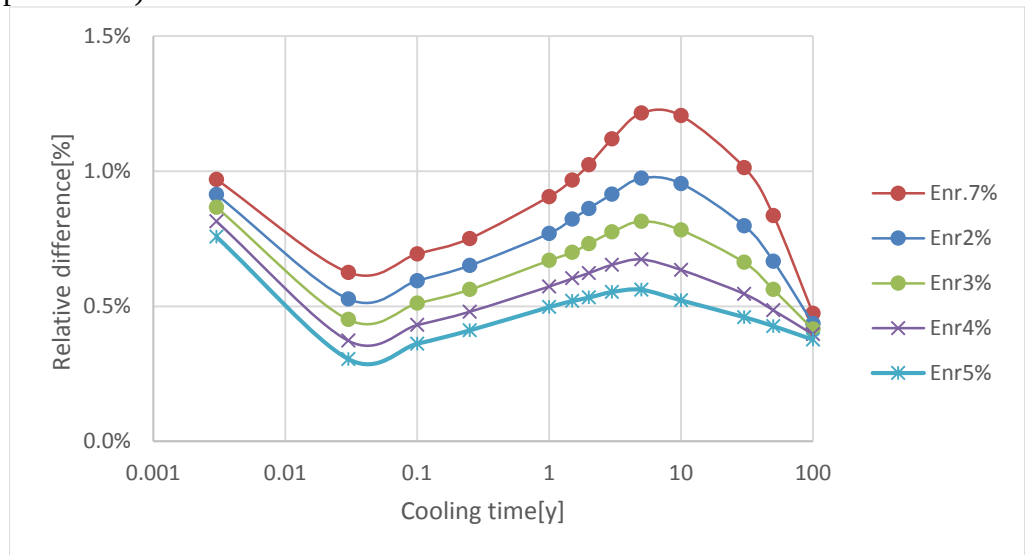

Figure 6: Relative differences of the ORIGAMI decay heat compared to the TRITON/NEWT results

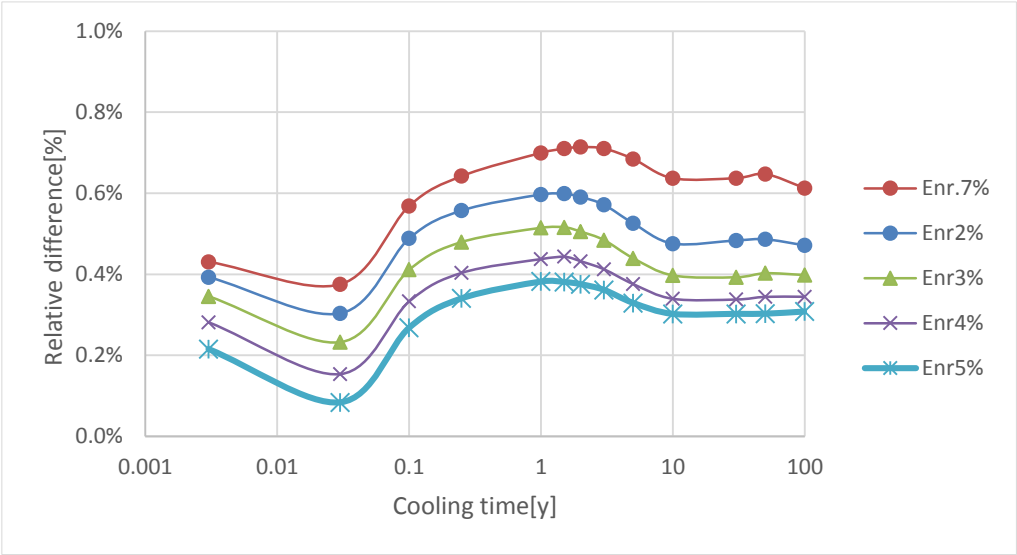

Figure 7: Relative differences of the ORIGAMI activities compared to the TRITON/NEWT results

Closer examination has revealed that the observed differences are due to the neutron capture reactions. Namely, in the FA non-fuel materials (e.g., structural materials) are also present. These materials contribute to the overall power production due to the energy produced by the neutron capture. For a given value of the total assembly power, this reduces the power from the fuel mass and thus slightly alter the fuel burnup and isotopics. The problem is that this contribution is case dependant and varies with enrichment, burnup, etc. The factor of the energy contributed to the fuel itself varies from 0.986 (enrichment 0.7 %) to the 0.997 (enrichment 5 %) for the presented cases. In order to show the dependence, a constant factor 0.996 was applied to reduce burnup steps in the .f33 files and specific power. Results are presented in Figure 8 and 9. Significant reduction in differences can be observed.

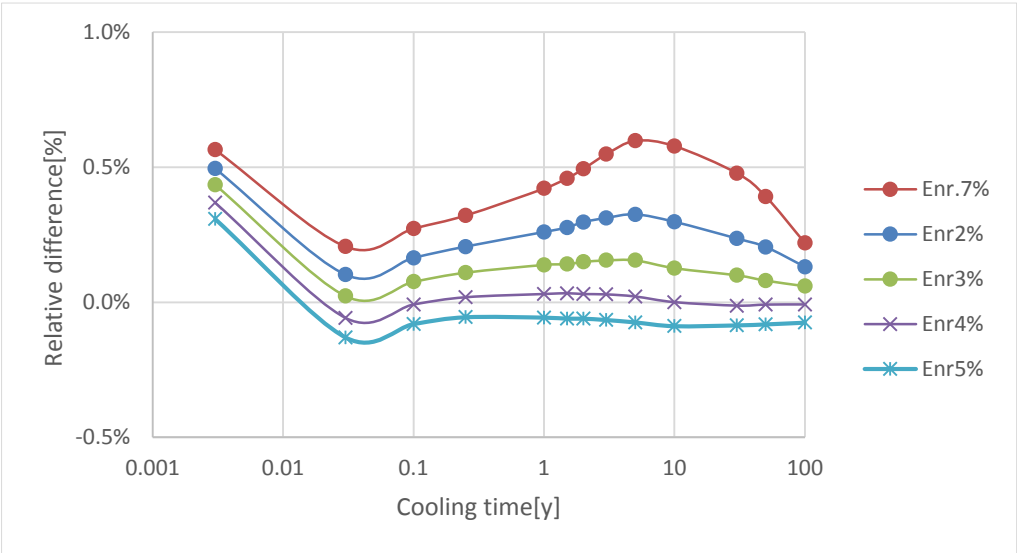

Figure 8: Relative differences of the ORIGAMI decay heat compared to the TRITON/NEWT results, neutron capture correction factor 0.996

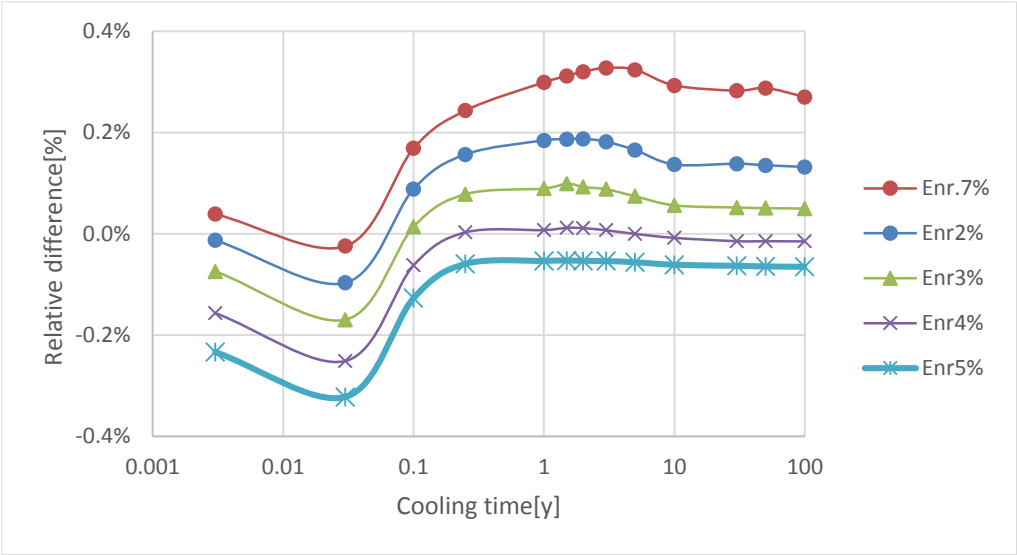

Figure 9: Relative differences of the ORIGAMI activities compared to the TRITON/NEWT results, neutron capture correction factor 0.996

4 SENSITIVITY ANALYSIS

Sensitivity analysis has been performed using the developed TRITON/NEWT model and parameters range defined in Table 1. Unfortunately, due to space limitation only enrichment and burnup are presented in the paper. Reference nominal case was constructed with parameters in the middle of the respective range, except for the enrichment and burnup, where 5 % and 60000 MWd/tU was judged to be more representative.

Sensitivity analysis defines response of the parameter variation on the observables and can be used to determine conservative approach, when explicit parameter consideration in the characterization process is not reasonable or viable, and define basic uncertainties due to propagation of uncertainties in each parameter and selected process. In the evaluation of the averaging process, non-linearity of the sensitivity coefficients should be examined also. Namely, calculations are usually performed at some averaged parameters value. It is not self-evident that such averaging process would yield also an averaged result.

4.1 Enrichment

Impact of different fuel enrichments on the decay heat is presented in Figure 10. Several enrichments have been compared to the reference 5 % enrichment. In the time period of 1 year – 50 years, lower enriched fuel produces higher decay heat. For the longer cooling times the situation is opposite. As already mentioned, in practice it is a usual procedure to calculate larger fuel areas with some average properties. But strictly speaking, if the fuel assembly is constituted of 2 geometrically equal regions, the one with 3 % enrichment and the other with 5 %, the average decay heat would not be equal to the decay heat of the fuel with an average 4 % enrichment. Relative errors of such averaging process are plotted in Figure 11. In the time period of 1 year – 30 years, averaging process leads to lower decay heat. For longer cooling times obtained decay heat will be higher.

Effect of the enrichment on the activity is presented in Figure 12 and Figure 13. Differences are pretty large, however, errors due to averaging process are smaller compared to the decay heat case.

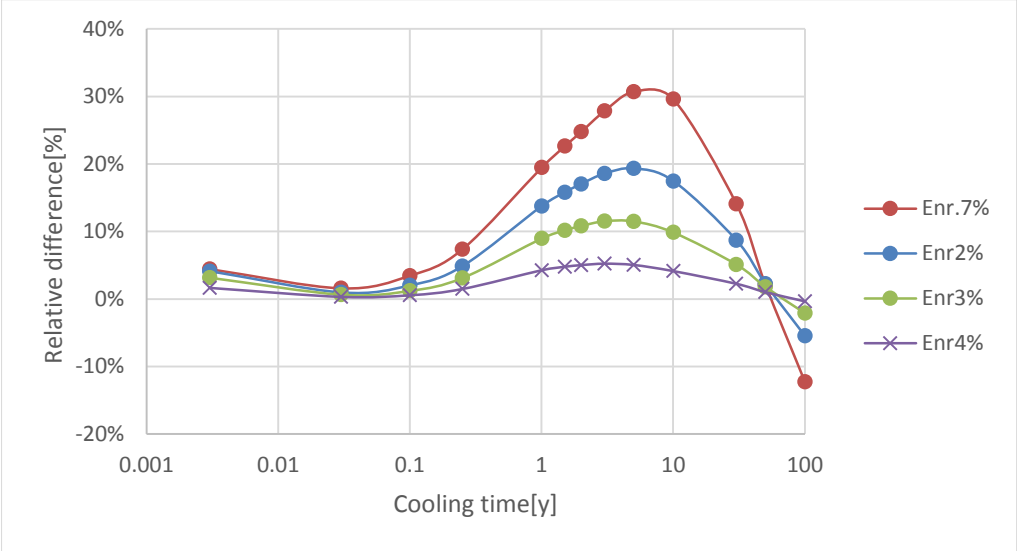

Figure 10: Impact of the fuel enrichment on the decay heat

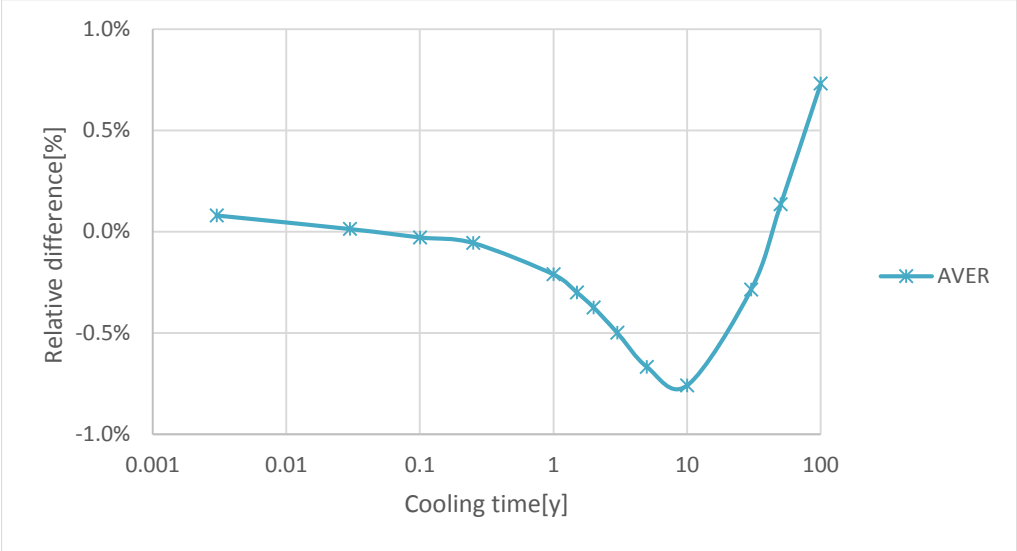

Figure 11: Impact of the ±1 % enrichment averaging on the decay heat.

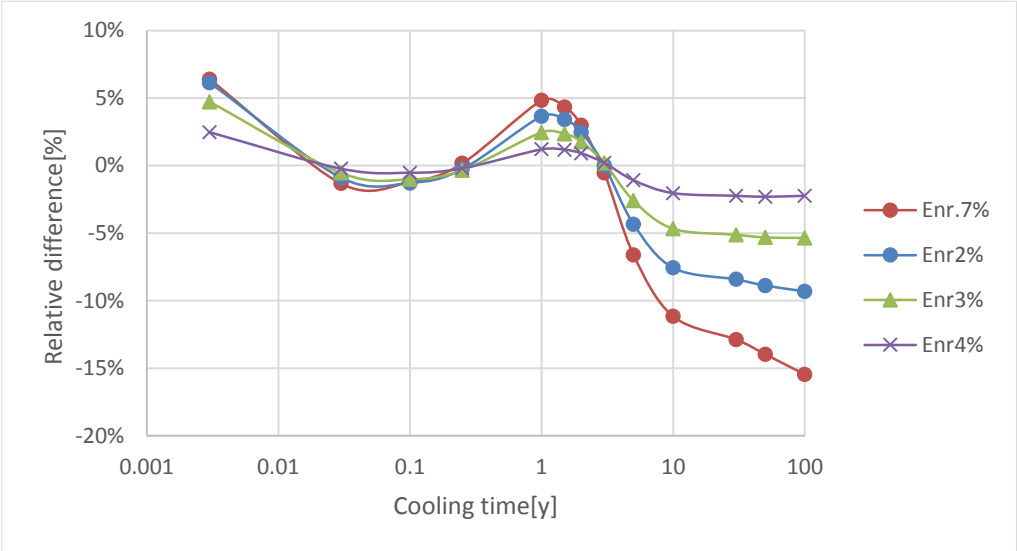

Figure 12: Impact of the fuel enrichment on the activity

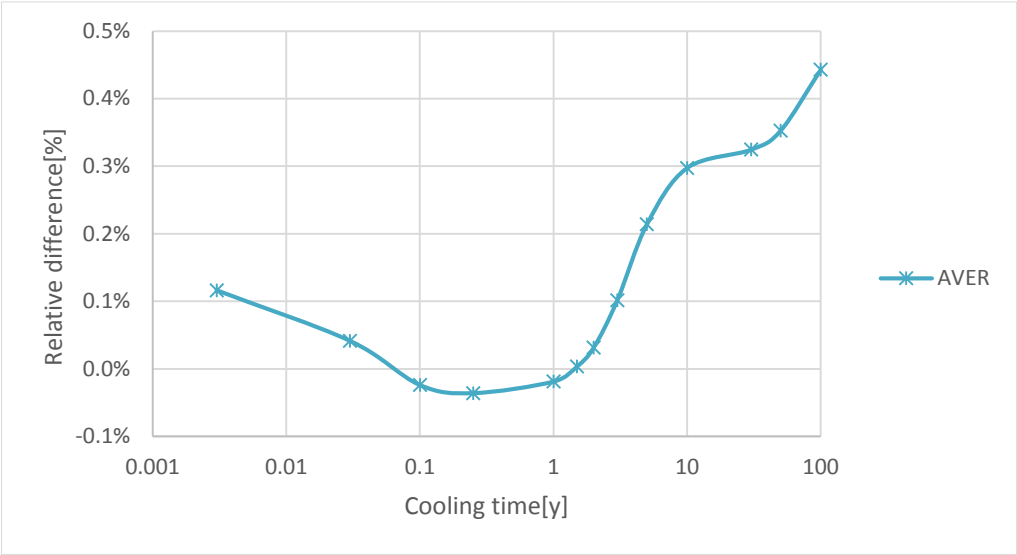

Figure 13: Impact of the ±1 % enrichment averaging on the activity

4.2 Burnup

Impact of the fuel burnup on the decay heat is presented in Figure 14. Two burnup cases 36000 MWd/tU and 48000 MWd/tU have been compared to the reference 60000 MWd/tU case. As expecting, decay heat is increasing with the fuel irradiation. Averaging process over considered ± 12000 MWd/tU burnup interval (denoted as AVER) is producing up to 2.5 % under prediction in the decay heat.

4.3 Other parameters

Impact of all other parameters given in Table 1 has been evaluated in the project. Due to programming and execution time constraints in ORIGAMI, only limited number of variables can be explicitly taken into account in the interpolation of the .f33 files. All others will have to be covered in some conservative manner. The final decision has yet to be made.

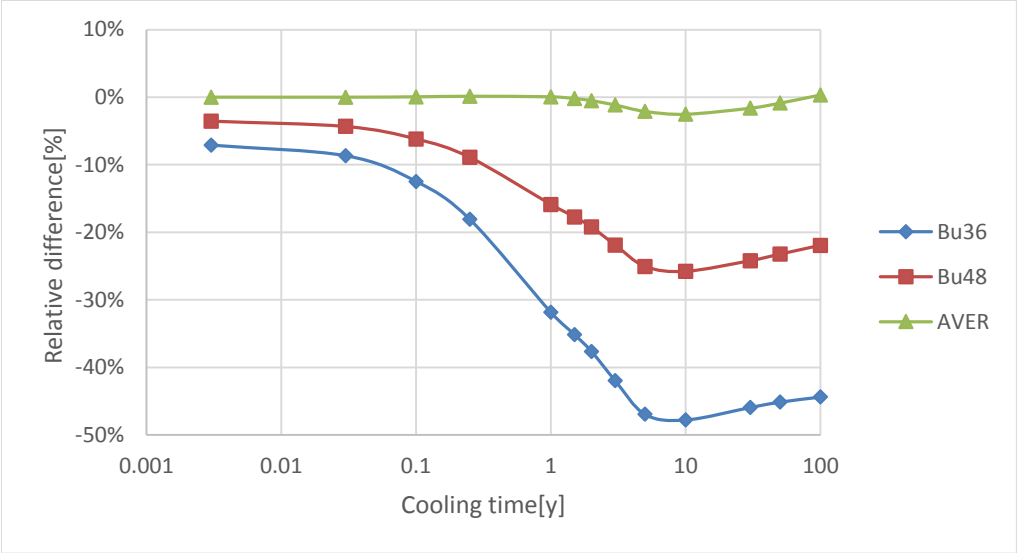

Figure 14: Impact of the fuel burnup on the decay heat

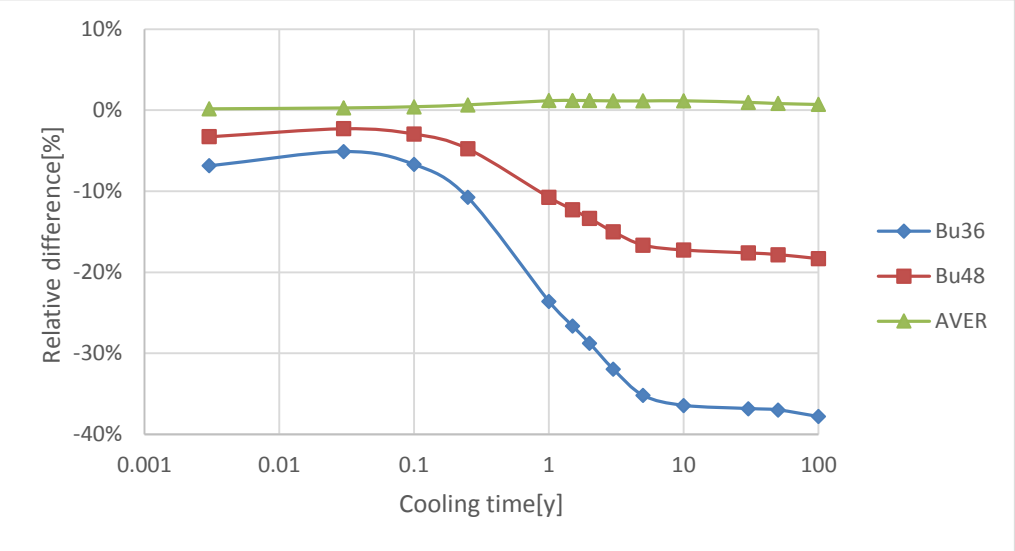

Figure 15: Impact of the fuel burnup on the activity

5 IMPACT OF AXIAL DATA SEGMENTATION

Performed sensitivity analysis has determined impact of individual parameters on the specific spent fuel observable. Unfortunately, averaging process yields to over prediction in some cases and under prediction in others, with possible opposite trends in different observables. It is very hard to identify the most important ones for all cases. The final result is a mixture of individual contributions. The most accurate approach would be to take all 10 axial regions explicitly. However, such approach is time consuming. It would take more than 3 years of the CPU time to calculate 1400 FAs with the TRITON/NEWT sequence. Even ORIGAMI would require several days of calculation. Some kind of homogenization is therefore necessary.

It seems natural to explore axial blankets, since there are large material heterogeneities present. To get a proper feeling, a few typical cases have been evaluated:

- A – standard FA without axial blankets (cycles 1-7)
- B – FA with natural uranium axial blankets (cycles 7-16)
- C – FA with annular axial blankets (from cycle 16 onward)

Results for the decay heat are presented in Figures 16-18 and activity in Figures 19-21. Firstly, we have checked homogenization of both axial end regions (1 and 10). The layers have the same enrichment and mass, and relatively similar all parameters except the temperature (density) of the moderator. Homogenization of both extreme layers is denoted by Ave(1,10). The differences between the homogenised calculation and the average values of the two individual layer calculations are very small and in all cases less than 0.1 %. The same procedure was performed for the central 8 layers (designation Ave(2-9)). Since the differences in the parameters are slightly larger here, the errors are also larger. Homogenization yields up to 0.7 % lower values in decay heat and up to 0.2 % higher values in activities. We estimate that the magnitude of these errors might be acceptable, given that homogenization brings us an 8-fold reduction in computational time.

Homogenization of all 10 layers (designation Ave(1-10)) gives much larger deviations. Decay heat is more than 2 % lower, while activities are up to 0.5 % higher for relevant cooling times. Such deviations may be too high, especially in light of the fact that the values for decay heat are underestimated (non-conservative). If we weigh the values obtained by homogenization of both extreme layers and homogenization of the central layers by mass weighting factors, the error is practically preserved (Ave(Ave(1-10), Ave(2-9)). Such an approach seems to be the most reasonable, since compared to the total homogenization of the fuel, the calculation time increases by only a factor of 2, and the errors are reduced by a factor of 3.

6 CONCLUSION

Activities performed at JSI to support NPP Krško Spent Fuel Dry Storage project are presented in this report. In the initial phase, FAR and FASLIB fuel assembly databases have been harmonized. Validated data from the FASLIB database served to analyse fuel manufacturing and irradiation data. The range of the following fuel parameters, needed to cover all spent fuel variations, was determined:

- Fuel enrichment
- Burnup
- Fuel temperature
- Moderator density
- Specific power
- Soluble boron concentration
- Number of BPR rods
- Number of IFBA rods

TRITON/NEWT calculation model has been validated by the comparison of the predictions obtained from the stochastic neutron transport code Serpent2. The focus was on the NPP Krško spent fuel decay heat, activity and neutral particle emissions for cooling times up to 50 years. Results obtained by both codes are in good agreement. For the time period of 1 year – 50 years decay heat differences are less than 1.3 %, in activity 1 %, while neutron and photon emissions are inside 5 % and 2 % respectively. The effect of the B1 approximation in the SCALE is less than 0.1 % in decay heat, less than 0.05 % in activity, less than 0.3 % in neutron emission and less than 0.2 % in photon emission. ORIGAMI model running on the TRITON/NEWT binary one-group reaction coefficients library (.f33 files) has been validated. The importance of the energy produced by the neutron capture has been stressed to reduced observed differences.

NPP Krško spent fuel sensitivity study has been accomplished based on the developed TRITON/NEWT model and fuel predetermined parameters range. It seems that fuel burnup and enrichment are the most influential parameters in the calculation of observables. In the evaluation of the averaging process, non-linearity of the sensitivity coefficients was examined also. The effect is mainly noticeable for the enrichment and burnup, where the decay heat prediction might be several % too low, if unsuitable segmentation is performed. A few FA cases based on the real data have been analysed to demonstrate reasonable axial discretization.

Planned activities are gradually coming to successful conclusion contributing to the safe and economical spent fuel dry storage on the NPP Krško site.

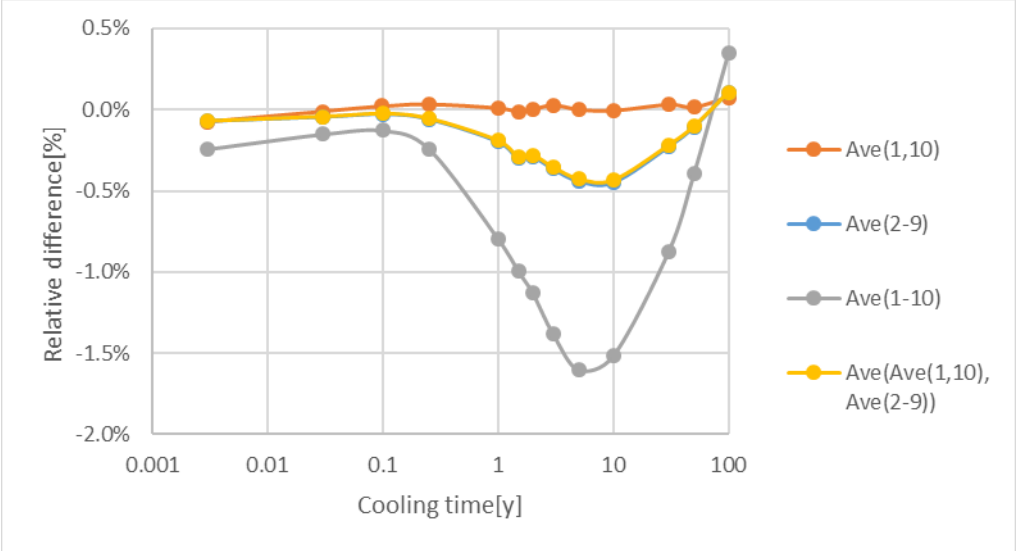

Figure 16: Impact of the homogenization on decay heat – case A

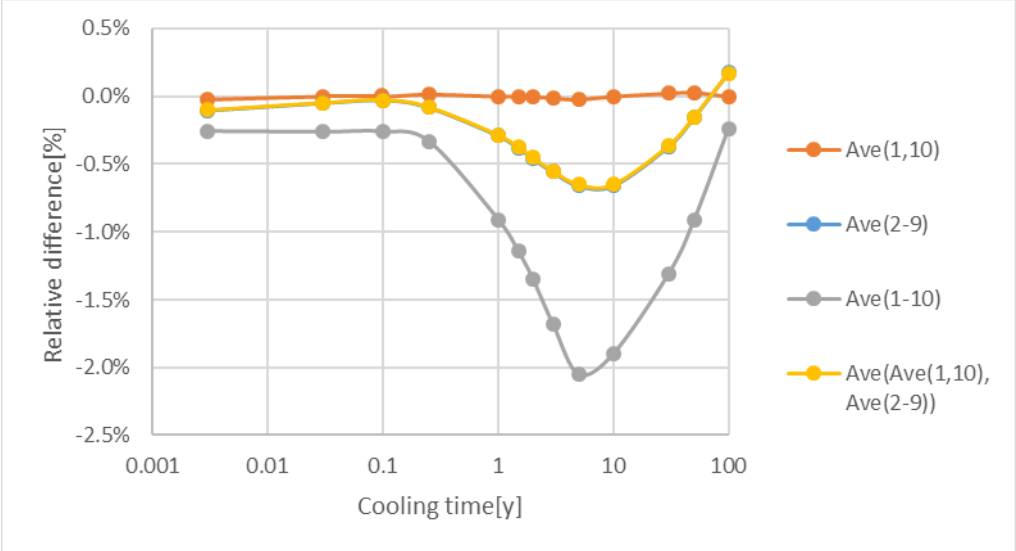

Figure 17: Impact of the homogenization on decay heat – case B

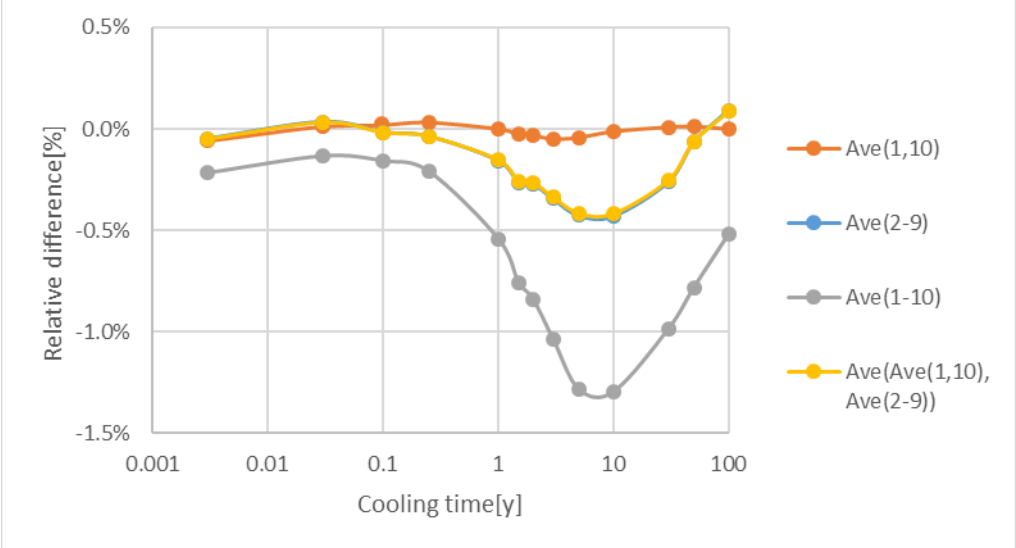

Figure 18: Impact of the homogenization on decay heat – case C

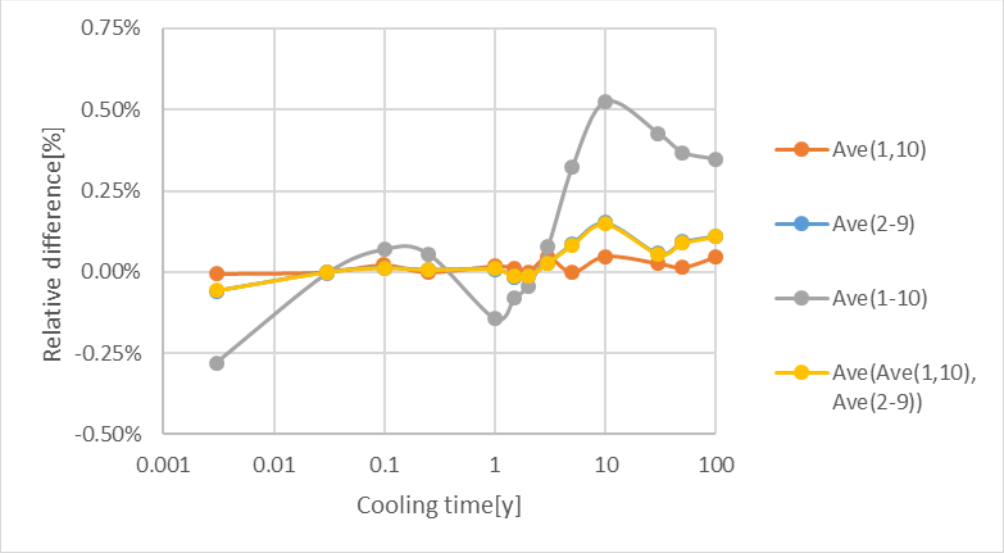

Figure 19: Impact of the homogenization on activity – case A

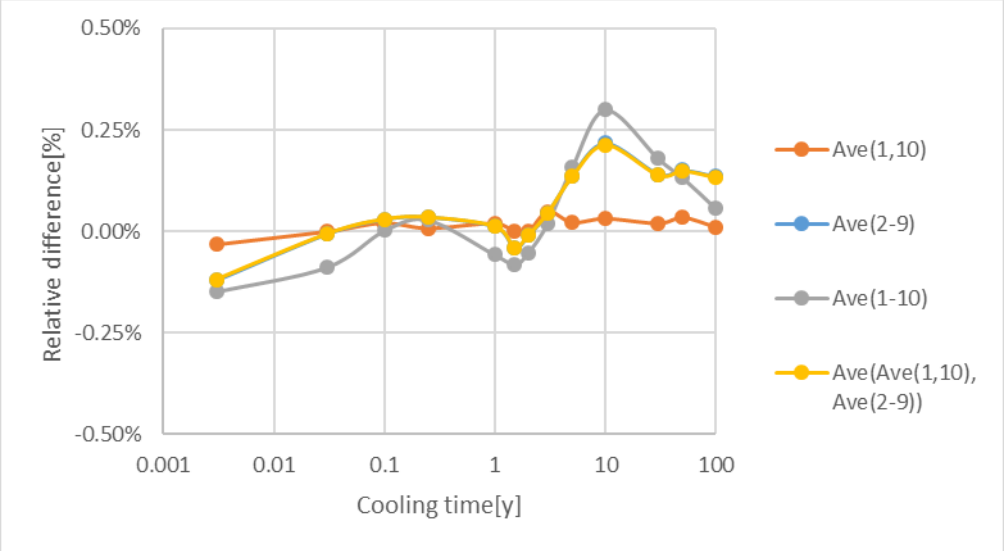

Figure 20: Impact of the homogenization on activity – case B

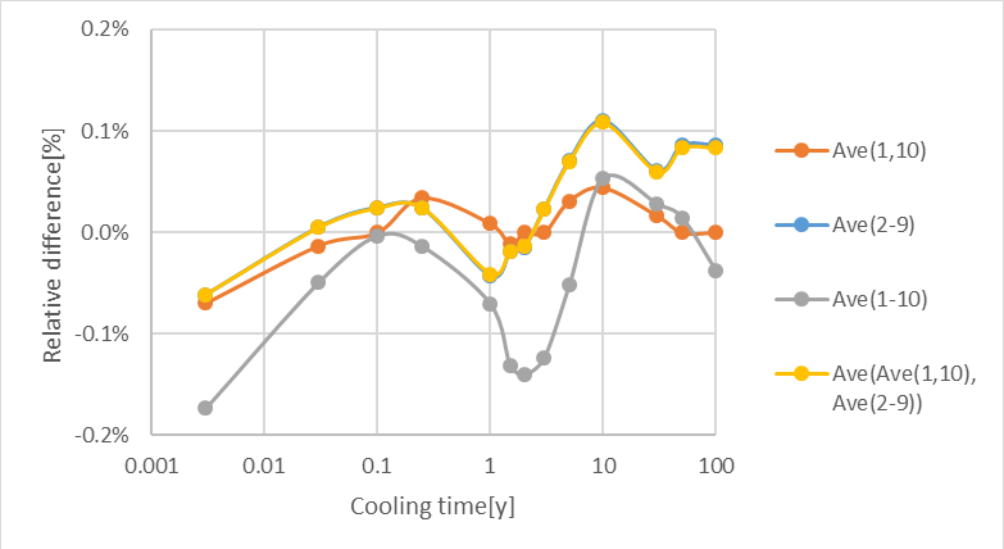

Figure 21: Impact of the homogenization on activity – case C

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