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Towards a Feasibility Study on BWR Burn-Up Credit

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Abstract

This paper presents our recent progress towards a feasibility study on BWR burn-up credit, the validation of the entire calculation chain and of all the involved codes. A sampling based uncertainty analysis was performed on a combined depletion and criticality calculation in order to obtain an integral overview of all effects involved and their impact on the reactivity of the spent fuel.

The analysis is done by first calculating nuclide inventories for a number of 2D slices of a generic fuel assembly representing different axial positions of parametrized burn-up profiles. In a second step, these nuclide inventories are inserted into a corresponding 3D model for criticality analysis of a generic transport cask and fuel storage pool. An empirical model is provided which reflects the physical and technical correlations and constraints in the uncertain parameters. For this task the SUnCISTT [1] code was used, coupling depletion calculations with the HELIOS [2] code to criticality calculations with KENO-V.a from the SCALE package [3].

In a preliminary step, uncertainty analyses were performed for the comparison between depletion calculations of BWR spent fuel isotopic composition experiments and their experimental values.

Introduction

The application of burn-up credit in the criticality assessment of transport casks for spent nuclear fuel has become state of the art in industry for pressurized water reactor (PWR) fuel in the last years. The phrase burn-up credit thereby refers to the method of taking advantage of the reduced reactivity of irradiated fuel in demonstrating the system's sub-criticality.

While the basic approach on how to apply burn-up credit in a system containing PWR fuel is widely agreed upon in the community, the situation is different in the case of boiling water reactor (BWR) fuel. The reason for this is the level of complexity that one is faced in BWR burn-up compared to the PWR case. This complexity originates from the radially and axially inhomogeneous spectral conditions in BWR fuel irradiation due to the density variations of the moderator, the complex geometrical design and also the control mechanisms in BWR operation, e.g. extensive use of control blades. All these aspects make it difficult to find simple criteria which provide bounding conditions and to define a practicable approach which allows for the validation of the computer codes involved.

In order to gain a comprehensive overview over the main parameters affecting the reactivity of spent BWR fuel in a storage situation (either a spent fuel pool or a transport cask) and their interactions, the work at hand applies a Monte-Carlo based uncertainty analysis to a combined depletion and criticality calculation. When compared to a single parameter variation study this approach has the advantage of also capturing common effects, i.e. additional effects of superposition in the variation of two parameters, which might be present in the problem. For this purpose the code SUnCISTT for Monte-Carlo based uncertainty analysis in fuel cycle applications is used. It has been developed at GRS over the last years, and has mainly been used for uncertainty analysis with respect to technical parameters in criticality calculations. It was enhanced by coupling depletion calculations with the HELIOS code to criticality calculations with KENO-V.a from the SCALE package.

S/U analysis using SUnCISTT

The GRS development SUnCISTT (Sensitivities and Uncertainties in Criticality Inventory and Source Term Tool) is a modular, easily extensible abstract interface program designed to perform Monte Carlo sampling based sensitivity and uncertainty (S/U) analyses. This analysis method uses repeated calculations of a given model with randomly varied input parameters to determine sensitivities and uncertainties of the model results. Originally, SUnCISTT was developed to determine uncertainties arising through manufacturing tolerances in assessments related to the nuclear fuel cycle. However, due to the abstract and modular code concept of SUnCISTT, its use is not restricted to these types of analyses. It is easy to extend in its functionalities and fields of applications due to its object oriented programming in Python3.

For the field of criticality safety, SUnCISTT couples different criticality and depletion codes to a statistical analysis tool, e.g. the well-established GRS tool SUSA [4]. Among these codes are the criticality sequences CSAS1, CSAS5, CSAS6 and T-NEWT of the SCALE package, developed at Oak Ridge National Lab [3], the particle transport code MCNP5, developed at Los Alamos National Lab [5], and the depletion codes OREST [6] and HELIOS. These depletion codes were selected mainly due to their low calculation time requirements. The depletion code OREST was developed by GRS for fast 1D pin cell depletion calculations, whereas HELIOS, developed by Studsvik, provides 2D depletion capabilities.

In Figure 1 the general scheme of the SUnCISTT work flow is shown. First a set of independent Monte Carlo samples need to be created by the user. Each sample consists of a randomly chosen value for each input parameter to be varied, where the choice of the values is based on the parameters probability density functions. Possible dependencies between input parameters can either be accounted for in the sample generation or using the user interface implemented in SUnCISTT. For each sample SUnCISTT creates a corresponding input file of the varied calculation model using a user defined and code-specific template file of the model to be analyzed, in which user defined keywords replace the nominal values of the uncertain parameters. Afterwards the input files need to be executed and SUnCISTT collects the calculated individual results from the generated output files. These results are stored in a separate file for further statistical analysis and visualization.

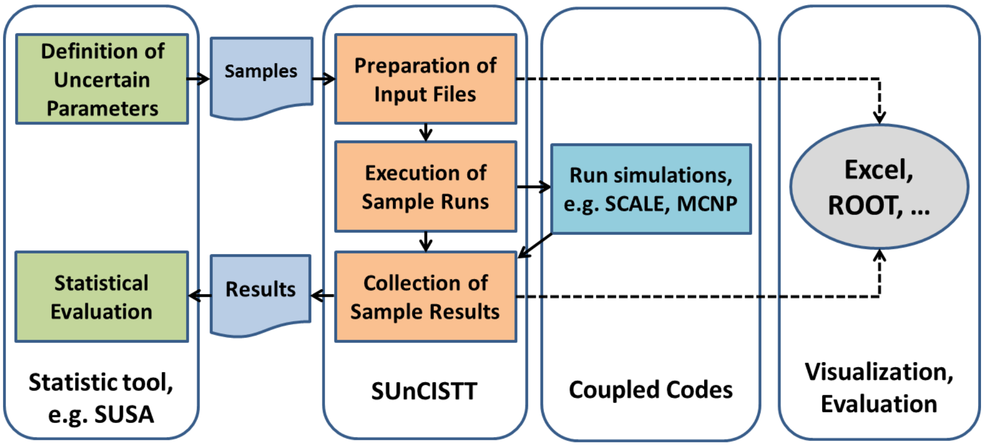


Figure 1: Sequence of a S/U analysis with SUnCISTT

In addition to S/U analyses of criticality calculations with a well-defined (e.g. unirradiated) fuel composition, SUnCISTT also features the capability to perform burn-up credit S/U analyses. In such a case the fuel composition is typically not predefined but has to be determined by a depletion calculation. Afterwards the calculated nuclide inventory has to be transferred to the calculation model for the subsequent criticality analysis. Consequently, SUnCISTT reproduces this two-stage process (Figure 2). In order to include 3D axial burn-up profiles of irradiated fuel assembles to the analysis, SUnCISTT was enhanced to handle not only one but an arbitrary number of template files. In this way, each axial burn-up zone can be modeled separately to take into account different axial burn-ups, moderator densities, partial length fuel rods, etc.

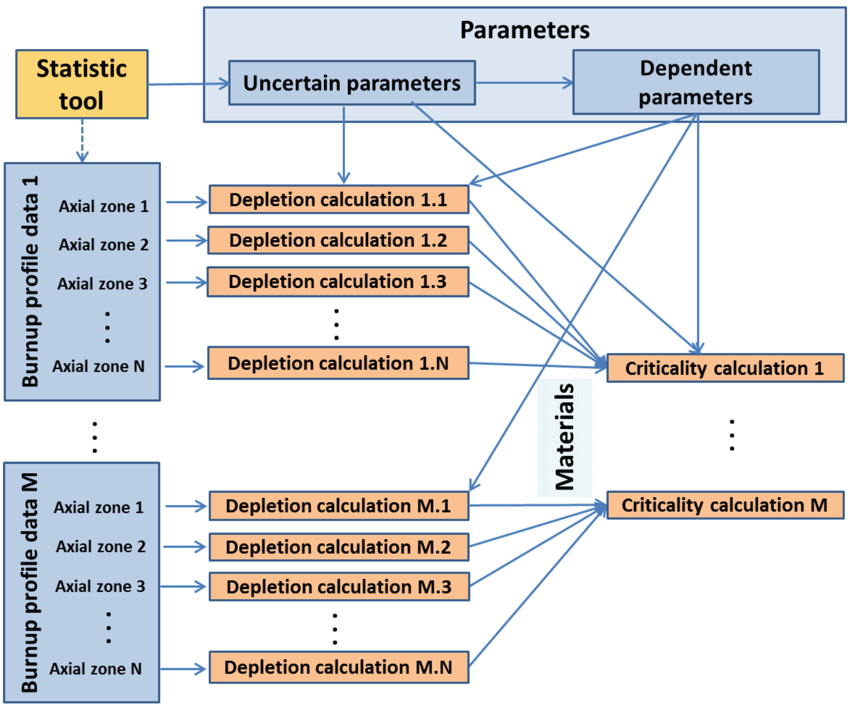


Figure 2: Two-step S/U analysis of a burn-up credit application case using SUnCISTT

Now, the user has to provide a set of independent Monte Carlo samples containing the chosen values of the varied input parameters for the models of both the depletion and the criticality calculation. In the first analysis stage SUnCISTT prepares the input files for the depletion calculations. These input files have to be executed. Afterwards SUnCISTT collects the results from the individual calculations, especially the calculated fuel compositions. If axial burn-up profiles were included, several depletion calculations need to be performed for each criticality sample run, one calculation for each axial burn-up zone. In the second stage of the analysis, SUnCISTT automatically creates an additional internal parameter and a corresponding keyword for each extracted fuel composition. These keywords can be used in the template file of the criticality model in order to consider the fuel compositions calculated in the first step. The keywords will be replaced by the corresponding fuel compositions during the input file creation. Afterwards the S/U analysis of the criticality part is performed as described above. The whole process is illustrated in Figure 2.

Uncertainty analysis of BWR spent fuel isotopic composition experiments

In a first step towards burn-up credit, an uncertainty analysis was performed to investigate the sensitivity of BWR spent fuel isotopic compositions on system parameters. Due to its relatively simple set-up the isotopic composition experiment of Fukushima-Daini-2 was chosen as exemplary model. However the information about uncertainties of current fuel assemblies, reactor data and burn-up histories is in general very scarce. For this experiment no such data could be found. Therefore, the uncertainties given for the isotopic composition sample F3F6 of the Swedish BWR Forsmark [7] are used for the S/U analysis of the samples from Fukushima-Daini-2. The used uncertainties are given in Table 1.

Table 1: Uncertainties for the isotopic composition sample F3F6 of the BWR Forsmark

Parameter	Label in Figure 3	Uncertainty
Outer cladding radius	**rCladOut**	**± 0,005 mm**
Fuel density	**rhoFuel**	**± 0,10 g/cm^3**
Void	**void**	**± 3,0 %**
Fuel temperature	**tempFuel**	**± 50° C**
Enrichment ^{235}U	**percU235**	**± 0,0005 %**
Burn-up	**Burn-up**	**± 3,5 %**

The uncertainty analysis was performed with SUnCISTT and includes the calculation of the Pearson correlation between the sampled uncertain input parameters and the resulting nuclide densities. All nuclides were considered, which are given in the chemical isotopic composition analysis. Figure 3 shows the results for sample SF98-05 originating from a non-gadolinium bearing fuel rod. The Pearson correlation coefficient can range from -1 (yellow) indicating a perfect negative correlation to 1 (dark red) indicating a perfect positive correlations. Values around zero (orange) indicate no correlation. The uncertainty of the shown values lies with 250 calculated samples around 0.1 for correlations around 0. The uncertainty approaches zero towards correlation coefficients of -1, respectively +1.

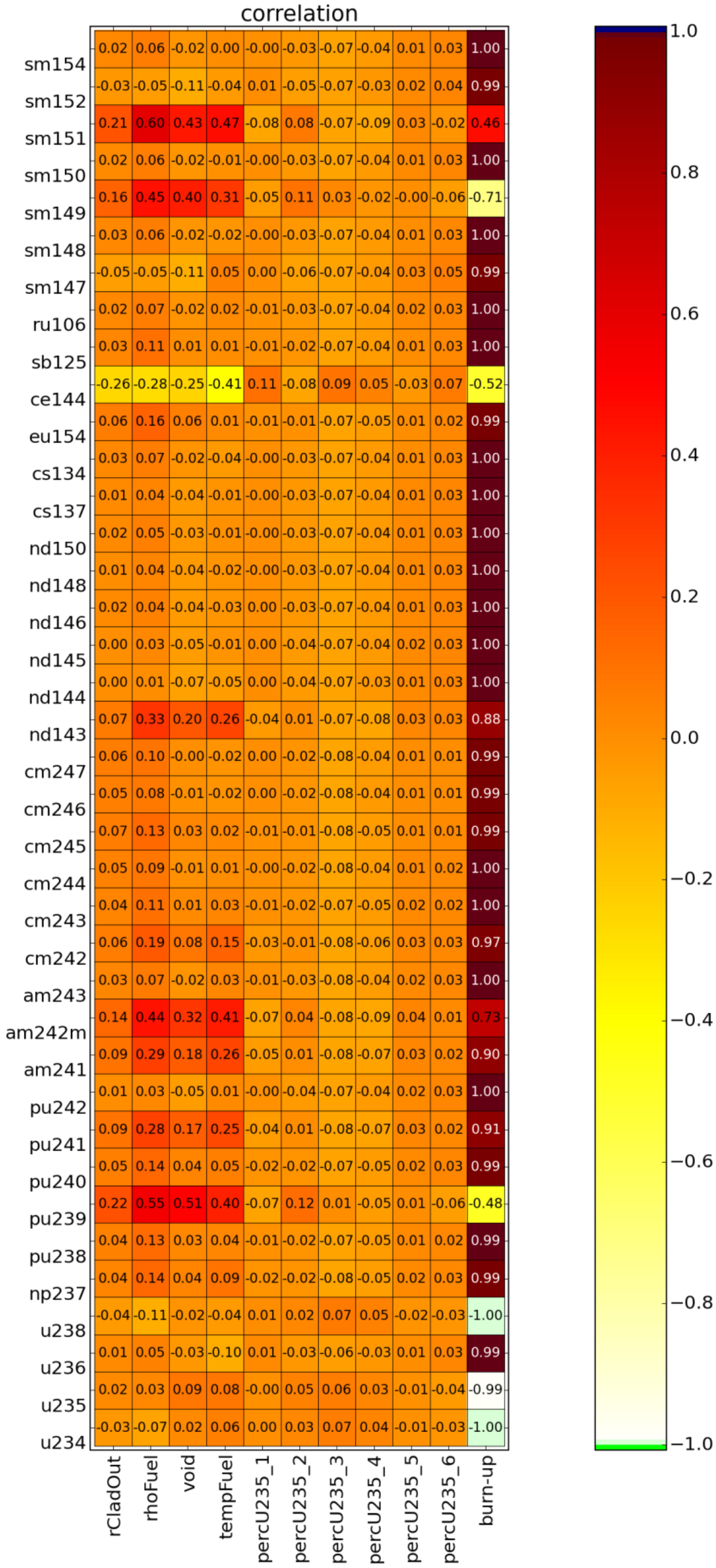
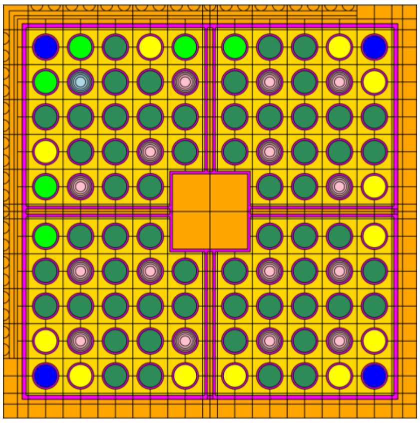


Figure 3: Correlation between nuclide densities and the varied parameters

The densities of almost all nuclides are sensitive solely to the burn-up of the sample. For a few nuclides ($^{149,151}Sm$, ^{144}Ce, ^{143}Nd, $^{241,242m}Am$, $^{239,241}Pu$), also those parameters are relevant which impact the neutron spectrum (fuel density, void, fuel temperature, and outer cladding radius).

Combined uncertainty analysis of depletion and criticality calculations

In a second step, a combined uncertainty analysis of depletion and criticality calculation for a generic set-up was performed. This paragraph describes the model used and presents some preliminary results. We have chosen a generic BWR fuel assembly (FA) roughly based on the SVEA-96 Optima3 design by Westinghouse [8, 9]. The criticality analysis was performed for a generic storage pool and a generic transport cask based on the CASTOR® V/52 design by GNS [10] with space for 52 FAs. Figure 4 shows the cross sections for the calculation models of the FA (HELIOS) and the transport cask (SCALE).



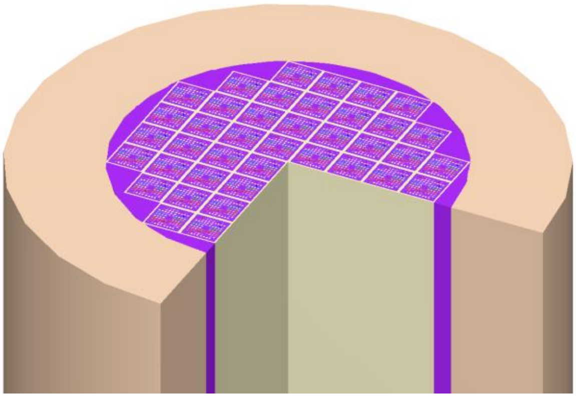


Figure 4: Calculation model of the FA and the transport cask

Development of the calculation chain

The irradiation conditions in BWRs are partly significantly different from PWRs. In order to take this into account, the used FA model contains the following features: five groups of FRs with variable enrichments, some FRs with reduced length (4:1/3, 8: 2/3), water channels between and within the FAs with non-boiling water.

Due to the high axial variability of geometric and control conditions, it was decided to create a depletion model with 24 axial zones and a full 2D geometry. For the uncertainty analysis 250 sample runs were calculated. The resulting huge number of calculations (6000 per analysis) calls for the use of a very fast 2D burn-up code, which was found in the deterministic code HELIOS v1.12. The model was verified by comparing calculations with TRITON from the SCALE package [3]. The criticality calculations were performed with KENO V from the SCALE package.

Generation of axial profiles

Due to the strong variability of the axial profiles (power, burn-up, density) a consistent model was developed for the fundamental quantities power, burn-up and void content. For this purpose, a dataset from BWR reactor core analysis calculations was analyzed, which provides profiles of power (mid of cycle), burn-up (end of cycle) and coolant density (mid of cycle). The three individual axial profiles were fitted by 4th grade polynomials, leading to 15 free parameters for each FA. By linear dependencies between polynomial parameters, the three variables could be correlated and the degrees of freedom could be reduced to two free parameters per cycle. These two parameters basically describe the average power level and the shape of the power profile of the assembly. The axial burn-up profile and the moderator density profile are then fixed automatically through the correlations described above. This gives in total ten free parameters for five cycles. The profiles for the actual calculations were sampled using empirical distributions separately for every cycle. Figure 5 shows the resulting profiles for the first cycle.

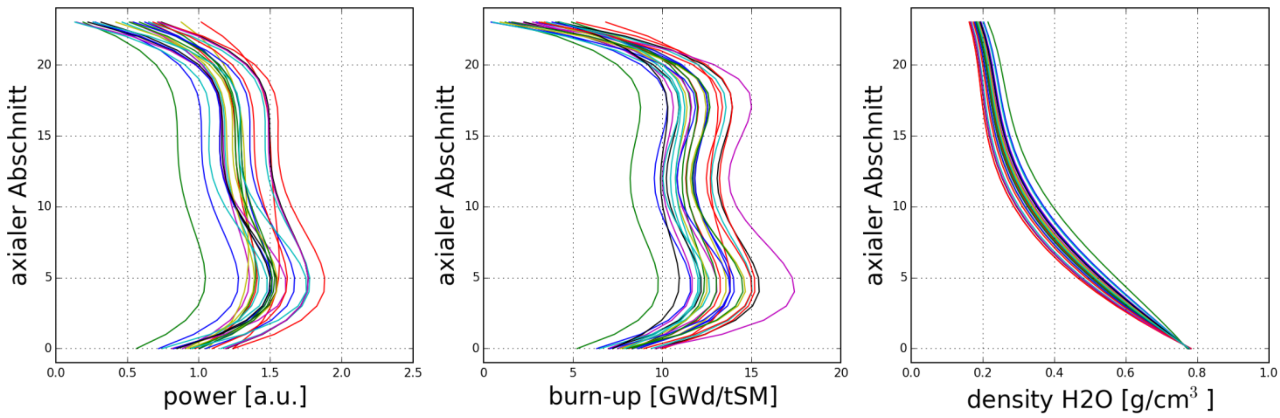


Figure 5: Sampled axial profiles for the first cycle

Preliminary results of combined uncertainty analysis of depletion and criticality calculation

The entire calculation chain was applied for several cases. First, 250 instances of the axial profiles of power, burn-up and void were created from the consistent model by Monte Carlo sampling. Using SUnCISTT, the corresponding values for each of the 24 axial zones were extracted and written into the 24 2D-HELIOS burn-up template files. Each of these templates was created via a Python3 script with four fuel rods (FR) with 1/3 length and eight FR with 2/3 length. The number densities of 44 nuclides relevant for burn-up credit were extracted from the 6,000 individual depletion calculations by SUnCISTT and were used in the subsequent criticality analysis. Utilizing the diagonal symmetry and considering the missing FRs in the upper part of the FA this sums up to a total of 12.8 million nuclides or 51 392 nuclides per sample run. All nuclides are transferred to the 250 sample runs of the 3D-SCALE criticality model of the transport cask. Its template input files are also generated via a Python3 script to ensure the correct bookkeeping of the huge amount of materials and calculations.

In a first step, a parametric study was performed to analyze the importance of the variation of the three parameters power, burn-up and void in each axial zone. For this purpose, the profiles were varied around the average values individually for each zone using a Gaussian distribution by +/- 10 % (sigma). This leads to discontinuous profiles and unphysical relations between the three values, but allows analyzing the influence on k_{eff} of each individual parameter in each individual zone. The results are shown in Figure 6 for the cycles one to five. The correlations between k_{eff} and the variations of power, burn-up and moderator density for each axial zone are divided in three rows, although they originate from one S/U analysis. The corresponding k_{eff} values and their uncertainties are shown in table 2.

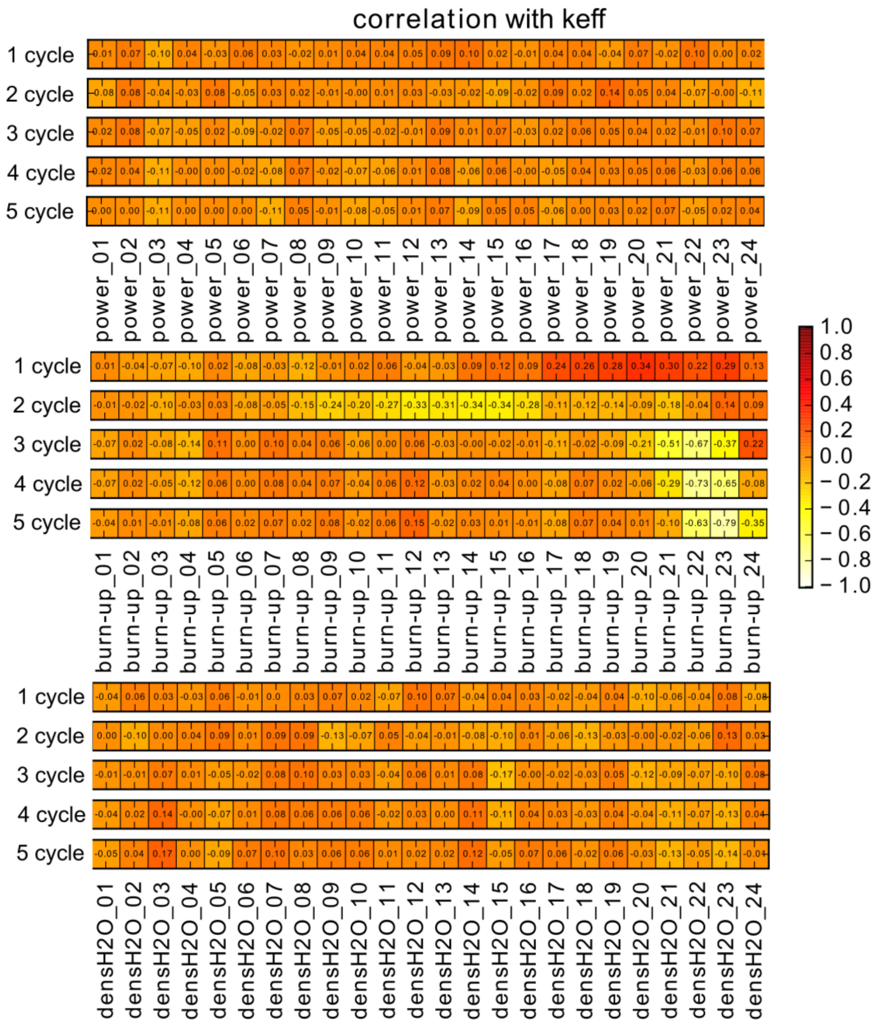


Figure 6: Correlation between the varied parameters and k_{eff}

Table 2: k_{eff} values of the transport cask applying the entire calculation chain

Case	$k_{eff} \pm \sigma_{keff}$
1 cycle	0.8242 ± 0.0023
2 cycles	0.7929 ± 0.0027
3 cycles	0.7510 ± 0.0056
4 cycles	0.7204 ± 0.0072
5 cycles	0.6829 ± 0.0094

Considering the correlation factors, this analysis shows that k_{eff} is almost exclusively sensitive to variations of the burn-up, and to a much lesser extent to the power or the moderator density. The correlations between these other parameters and k_{eff} lie within the uncertainty of the correlation factor ($\sigma_{corr} \sim 0.1$). After one cycle (Figure 6, first line) the upper third of the FA is relevant for k_{eff} with a positive correlation (red). Since at this burn-up of roughly 9.5 GWd/MTU the gadolinium peak is not yet reached, this result is expected. For the expected final burn-up of roughly 55 GWd/MTU after five cycles (Figure 6, bottom line) only the three top-most zones are relevant for the criticality of the stored or transported FA with a negative correlation (yellow), if each zone is varied individually. This is expected, since higher burn-up leads to less fissile material, leading to smaller k_{eff}. One can see that after the gadolinium peak with increasing burn-up the relevant zones move more and more to the top of the fuel assembly.

The next step is to apply the consistent model of the axial profiles introduced above to the combined uncertainty analysis of depletion and criticality calculation. In this case only the ten independent profile parameters are varied. The sampled profiles are then used for the burn-up calculations. Preliminary analyses show that the first parameter (describing average power and burn-up) is slightly more relevant than the second one (describing the shape of the profile), but that all cycles are more or less equally important for the value of k_{eff} in the transport cask. More and deeper analyses will follow in this area.

Conclusions

In the paper at hand, our recent work towards a feasibility study on BWR burn-up credit is presented. Sampling based uncertainty analyses were performed on a combined depletion and criticality calculation. For this purpose generic 3D models of a transport cask and a fuel storage pool filled with SVEA-96 Optima3 like BWR fuel assemblies were created. In order to take into account the axial profiles of power, burn-up and moderator density, the fuel assemblies were divided into 24 axial burn-up zones. The values for each burn-up zone were parametrized using a consistent model which is based on a dataset of BWR core analysis calculations. The goal of these analyses was to obtain an integral overview of the main effects involved and their impact on the reactivity of the spent BWR fuel.

First results were obtained. For the analyzed generic model, k_{eff} is almost exclusively sensitive to the burn-up uncertainty. While the correlation is positive before the gadolinium peak, it becomes negative after the peak. The sensitive axial region moves more and more towards the top of the FA for increasing burn-up. The application of the entire consistent model for five cycles shows that the final k_{eff} is sensitive to all cycles.

The analysis will be continued by an extensive survey for different scenarios, also around the gadolinium peak. The impact of the use of control blades will be examined in the following. On the basis of these analyses the feasibility of BWR burn-up credit is investigated. An estimate will be given, if the available data is sufficient or has to be improved or extended. This will include information, in which area improvements most likely will have the largest impact on BWR burn-up credit.

Acknowledgments

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