BOUNDING APPROACH IN BUC IMPLEMENTATION IN POOL AT VVER-440

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ABSTRACT

As new fuel designs (with higher enrichment) are introduced, spent fuel storage facilities might not fulfill criticality safety criteria when using fresh fuel approach to the criticality analyses. Since optimum moderation conditions evaluation even in wet storage systems is required by some regulatory bodies, any credit for soluble boron cannot be taken. Thus, the only suitable way to prove subcriticality of the given spent fuel system with higher enriched/burnt fuel is burnup credit implementation.

This paper outlines burnup credit implementation methodology as demonstrated on criticality evaluation of VVER-440 reactor pool at NPP Dukovany. Operational history effects, isotopic set choice, as well as computational issues (SCALE 4.4a was used both for depletion and criticality calculations) are discussed, maintaining strictly conservative approach.

Bounding approach in operational history treatment was carefully examined. Criticality evaluation using selection of (as expected) conservative values of operational parameters (specific power, fuel and moderator temperatures, boron content in moderator) was compared to criticality evaluation of real fuel assemblies from the NPP database. Therefore, bounding approach was justified and it was shown that it is not excessively conservative.

INTRODUCTION

Burnup credit (BUC) – what and why

In order to increase spent fuel storage/transport systems capacity or to be able to load more enriched fuel types, fresh fuel approach should be replaced by another method of criticality evaluation, which is not so excessively conservative.

When partial boron credit (PBC) cannot be implemented (for example when low density optimum moderation conditions have to be evaluated), credit for fuel burnup has to be taken. This means that the considered amount of fissile material (mostly U-235) is reduced according to fuel depletion. Furthermore, buildup of fission products (which are more or less strong neutron absorbers) can be taken into account.

In BUC implementation one has to be much more careful in selection of all parameters of the model. All choices have to be strictly conservative, because there is no more that enormous margin formerly caused by fresh fuel approach. Thus, guidelines for conservative, yet not excessively conservative, choices of parameters will be formulated.

Examined system and used tools

BUC implementation was performed on upper rack of the reactor pool at NPP Dukovany (VVER-440); there are two racks in the pool. The lower one is compacted, but this works examines the upper one with 225 mm pitch. The fuel assemblies are held in place only with three thin steel grids, so there is almost no absorption material and the whole system is highly overmoderated.

Despite of the larger fuel assembly pitch, the upper rack is more vulnerable to become critical than to lower because of the lack of absorption materials. Thus, for newer fuel designs, fresh fuel approach is insufficient and credit for fuel burnup has to be taken.

For depletion and criticality calculations, SCALE 4.4a code complex was used, mainly for its ability to perform 3D Monte Carlo criticality calculations and to generate case-dependent nuclear data libraries during depletion calculations.

BUC IMPLEMENTATION

Choices to be made

During the BUC implementation, suitable model of the examined system has to be chosen. This includes:

- geometrical model (conservative approximations have to be made, because the system can never be described precisely)
- set of isotopes that are to be included into calculations (not all existing nuclides can – nor should – be taken into account)
- calculational parameters (of course, this stage is very different depending on the code used, but e.g. for Monte Carlo codes there are some general guidelines)
- operational history (usually, exact history of each fuel assembly in the system is not known and therefore some estimate has to be made)
- system parameters (like temperature, manufacturing tolerances etc.)

Isotope sets

Hundreds of nuclides are present in spent fuel. It is not suitable to include all of them into calculations – one problem is enormous increment in time consumption of such calculation and secondly, such complexity can become a serious source of errors.

First important choice is whether to include fission products or only actinides into calculations. Including only actinides (this is, considering only fuel depletion, not build-up of absorbers) is more conservative and easier to implement, but it was shown (see [2]) that omission of fission products leads to incorrect description of physical reality (e.g. inverse dependence of reactivity on some operational parameters). So it is probably better to take fission products into account.

[1] gives simple, but very important rules for isotope choice:

- no isotopes, which are not fixed in the fuel matrix (like gases or volatile elements) should be included
- short-lived isotopes (with half-life lower than a year) should not be included (their consideration introduces rise of k_{eff} in first days after reactor shutdown, so calculation cannot be performed with zero cooling time)
- properties of all included isotopes (especially cross sections and decay schemes) have to be well documented
- for all included isotopes, validation of used code (+nuclear data library) has to be performed

According to those rules, internationally accepted set of 27 isotopes was created. These are the most important nuclides in sense of criticality.

The set, which was also used for calculations in this work, consist of:

- Actinides: U (234, 235, 236, 238), Pu (238, 239, 240, 241, 242), Am (241, 243), Np-237
- Fission products: Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd (143, 145), Sm (147,149,150, 151, 152), Eu-153, Gd-155

It should be noted that validation of commonly used codes for VVER fuel was never correctly performed. No well-documented and comprehensive post-irradiation experiments were conducted, so results of validation for PWR fuel were used. Although PWR and VVER reactors have quite different neutron spectra (and therefore reliability of the codes can be different in those two areas), there is no other option until necessary experiments are performed.

Moreover, to examine and demonstrate effect of different isotope choices, calculations for several isotope sets were performed. Figure 1 shows k_{eff} development in time for 40GWd/tU burned fuel for various sets. When more isotopes are included, conservativeness can be reduced by another few percents, but those additional isotopes were never validated. Furthermore, two sets (which include those short-lived nuclides, mainly xenon) show rise of k_{eff} in first few days.

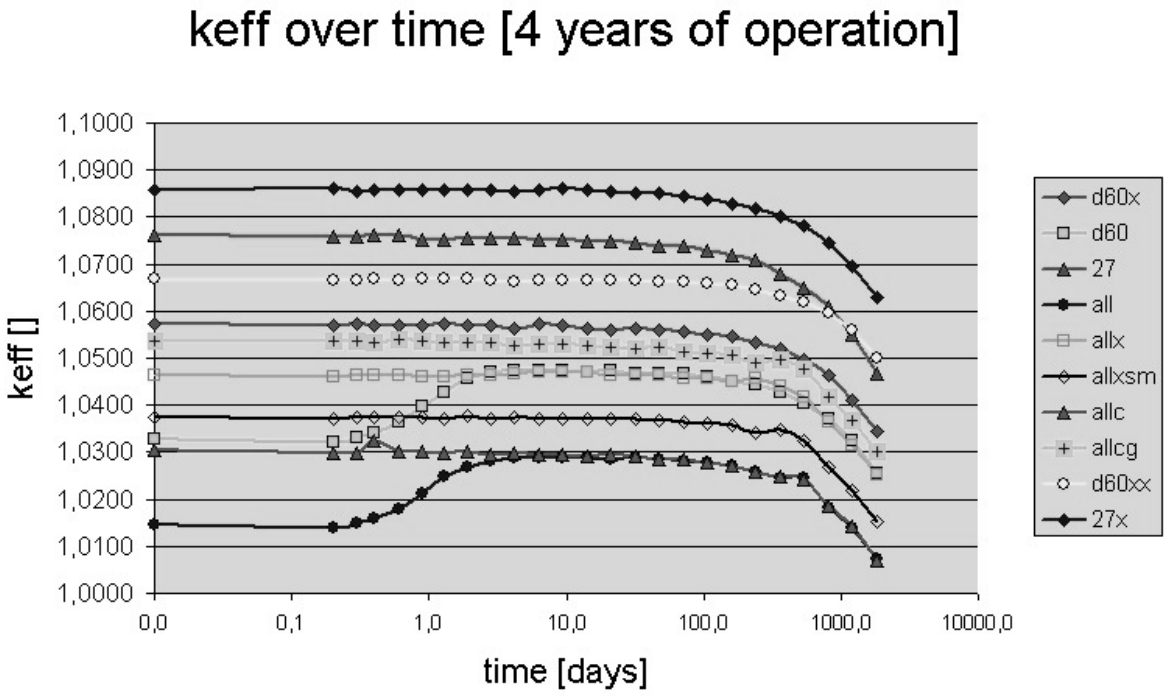

Figure 1: effect of different isotope choices.

Calculational parameters

Using SCALE, two parameters are important: adequate sampling for Monte Carlo calculations (e.g. sufficient number of neutron generations and generation size) and sufficient frequency of neutron data library recalculation during depletion.

The library can be recalculated several times during one depletion cycle (one operational period). In [5], dependence of resulting k_{eff} on this frequency was calculated for various fuel burnups and cooling times; in figure 2 there is an example of such a calculation. It follows that four recalculations per cycle is a good choice.

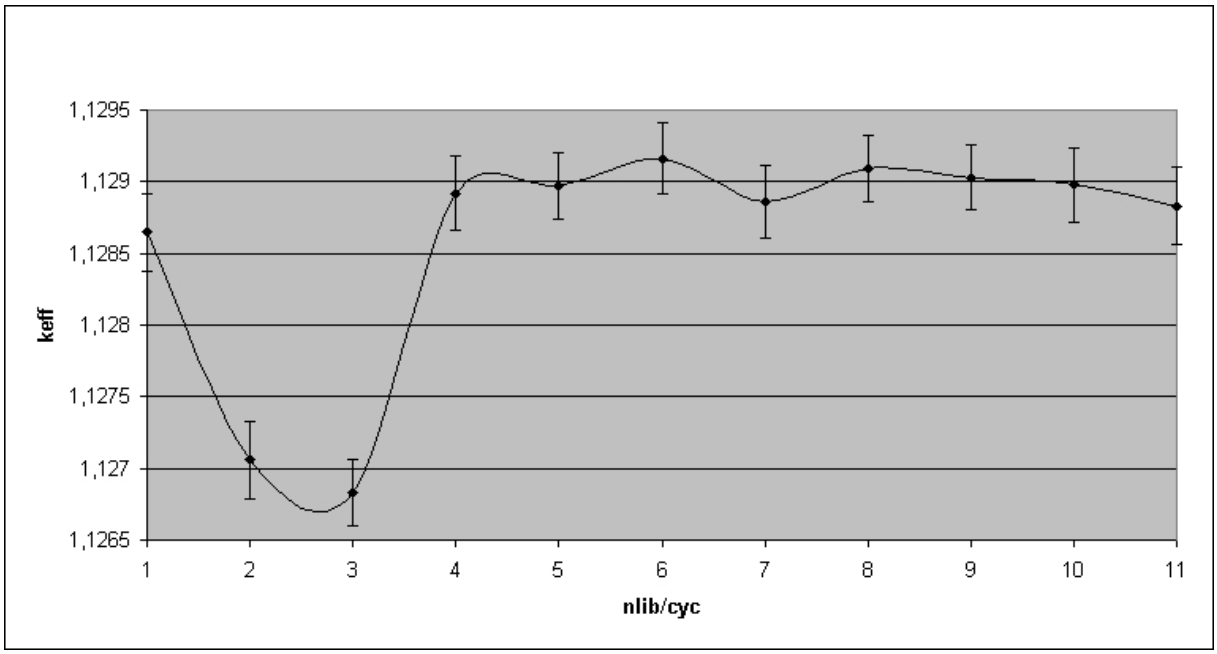

Figure 2: effect of a calculational parameter

Of course, when other codes are used, other parameters are involved. It is just important to understand that dependence of all possible parameters has to by examined for every case.

More general issue is population size in Monte Carlo codes. Inadequate sampling does not only introduce statistic error, but when the number of neutrons per generation (which represents precision of the approximation) is insufficient, the estimate can be biased. Figure 3 shows that for different sampling parameters the results can converge to different eigenvalues (and not only to the real k_{eff}). Thus, tests of sampling sufficiency should be conducted for each case (sampling requirements grow quickly with increasing geometrical complexity of the problem).

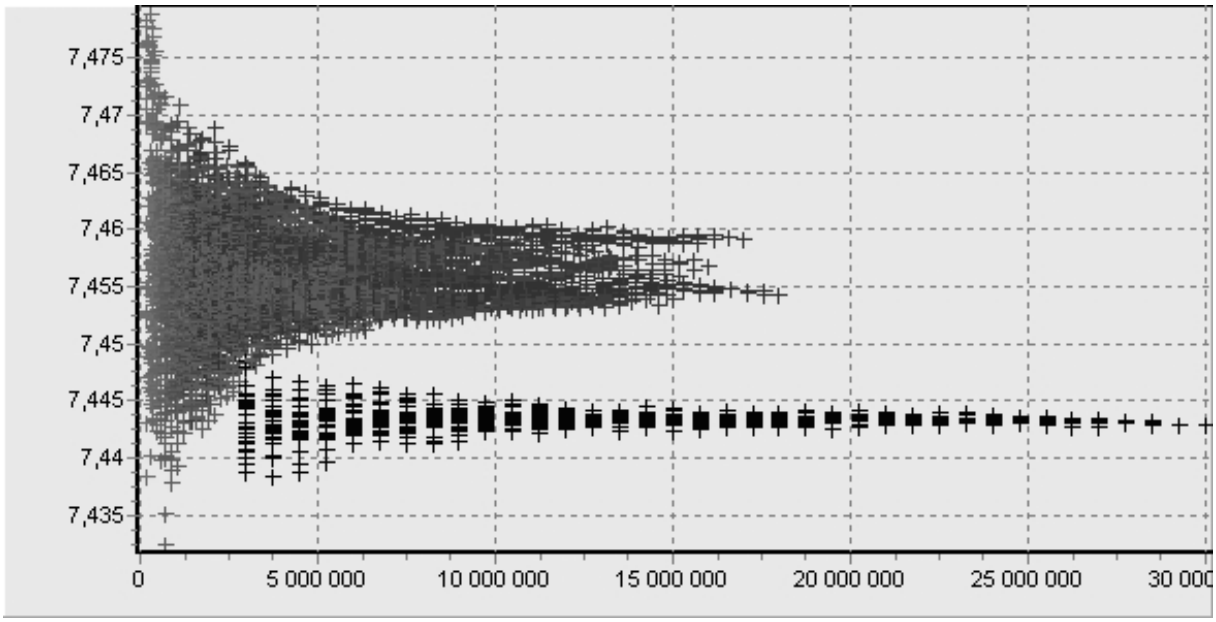

Figure 3: convergence of Monte Carlo code to different eigenvalues.

Influences of operational history

All operational parameters have impact on reactivity of the spent fuel. The main influences are: fuel and moderator temperature, boron content in moderator, specific power and downtimes.

Dependence on all of these parameters is well documented for both PWR and VVER application areas (see [1] and [3]). An example of such dependence can be found in figure 4.

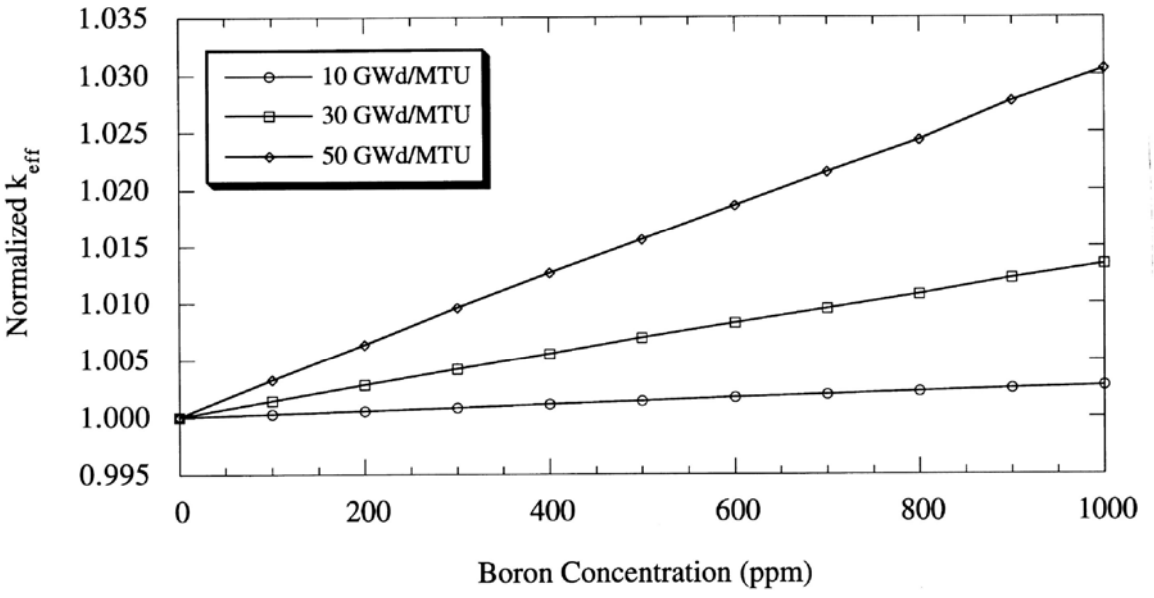

Figure 4: how boron concentrations affects resulting reactivity. From [1]

Bounding approach

Usually, particular operational history data are not available for each fuel assembly present in the system. Thus a reasonable (and conservative) estimate of depletion history has to be found.

Formerly ‘best-estimate’ approach was used, but nowadays it seems to be better (and of course more acceptable for the regulatory bodies) to follow strictly conservative approach. Therefore it is not possible to use average values of operational history parameters, but the ones which lead to higher calculated k_{eff} of the spent fuel. Selection of these limiting values is called bounding approach.

For individual parameters, it is possible to choose extreme possible values of parameters (minimum or maximum, as follows from known dependence of spent fuel reactivity on those parameters) using technical specifications or – if it is long enough – operational history records of the reactor.

But a question comes up – do not the parameters interact among each other in such way that this choice would not be conservative? To examine this problem, database of all fuel assemblies of one design (initial average enrichment 3.82%, no burnable poisons) at NPP Dukovany was taken and depletion+criticality calculations were performed for each single assembly. Afterwards, bounding parameter values were selected and the calculation was performed for such fictional fuel assembly.

Results of this examination were very pleasing and can be found in figure 5. The cross signs represent individual real fuel assemblies from the database and the line shows k_{eff} of the bounding fuel assembly.

Thus, bounding approach was found to be the very best one – it models the ‘worst possible’ (having highest k_{eff}) fuel assembly and therefore it is not excessively conservative, but still is conservative. So it is recommended to use this approach for operational history approximation.

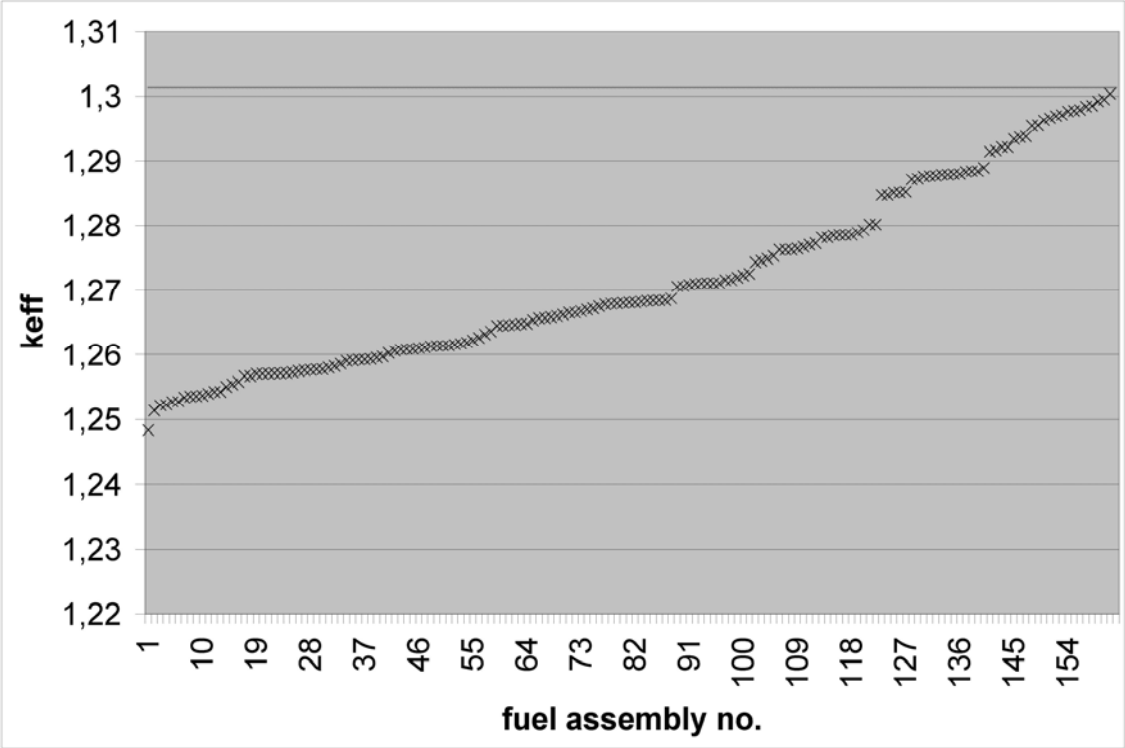

Figure 5: Bounding approach evaluation.

Pool temperature effect

As an example of unexpected effect of various parameter changes, dependence of k_{eff} on pool temperature is presented. It would be natural to assume that temperature reactivity coefficient is negative, but because of overmoderation of the system, the coefficient is positive and temperature causes rise of k_{eff}, as it can be seen in figure 6. Therefore, calculations have to be performed at the highest possible temperature (some regulatory bodies accept the maximum value as given by technical specifications of the pool, some require 100 ^{o}C).

This case gives a clear warning that effect of all parameter changes (including manufacturing tolerances) should be examined by careful calculation, not only by intuition.

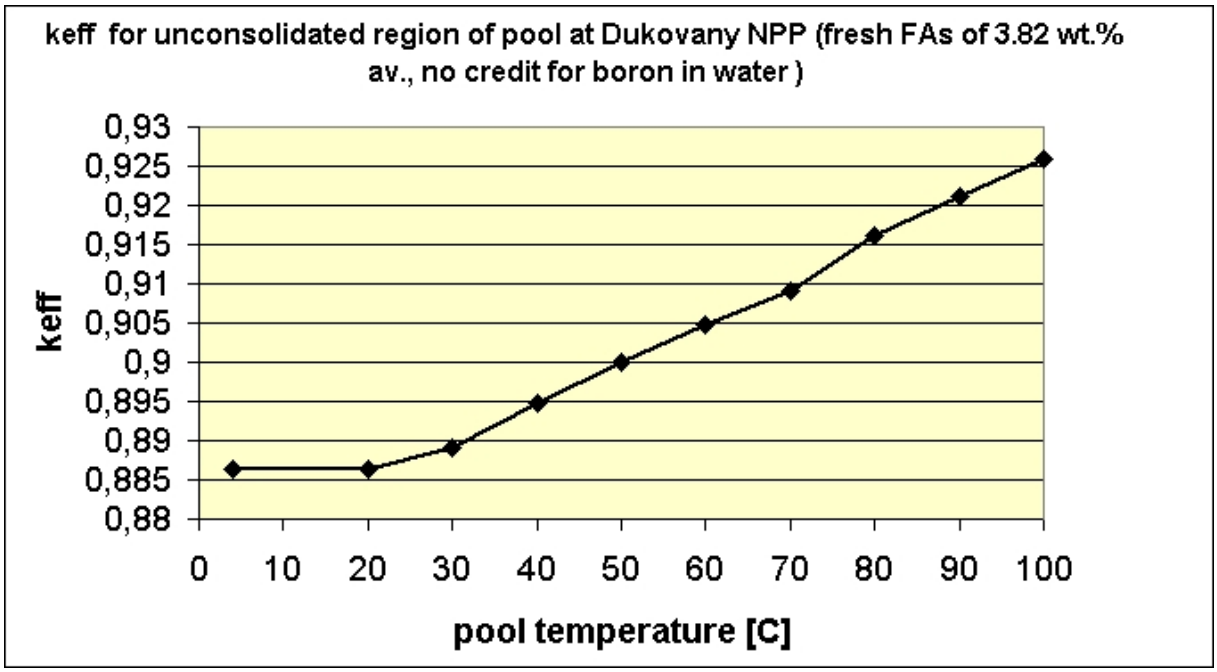

Figure 6: Effect of fuel temperature on k_{eff}. From [4]

CONCLUSIONS

In this work, guideline for practical BUC implementation was presented. General hints of conservative choices of all modeling and calculational parameters were given.

It was also found that operational history can be approximated by bounding approach, i.e. by estimating operational parameters by their limiting values (based on known dependencies of reactivity on those parameters).

It was also pointed out that several influences have to be carefully examined for each case (particularly Monte Carlo parameters, effect of system temperature, manufacturing tolerances).

NOMENCLATURE

BUC	burnup credit
k_{eff}	k-effective, effective multiplication coeficient
PBC	partial boron credit

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