EFFECT OF NUCLEAR DATA ON THE DNBR PREDICTION WITH SUBCHANNEL CODE COBRA-TF

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ABSTRACT

In this work, the effect of nuclear data uncertainties on the DNBR predictions is presented on a selected PWR fuel assembly with 15×15 rods. Assessment of several CHF correlations and CHF look-up tables in propagating pin power uncertainty to the DNBR predictions by each of these correlations is performed. Pin power map uncertainty is generated in combination with CASMO using the SHARKX tool which generates a set of random nuclear data libraries. These nuclear data uncertainties derived previously in [1] and characterized by skewed PDF distribution, which is applied in this work. Pin-wise DNBR analysis is performed with subchannel code COBRA-TF (CTF). The pin power uncertainty is applied to the variation of the relative radial rod power coefficient with 481 samples. This variation of relative rod power redistributes the fixed total FA power among rods, which is leading to variation of DNBR value for the selected rod, and this variation is captured in this work. The main outcome is that the standard deviation is increased around two times in comparison to the initial ppp distribution. Conclusions of this work can be important in the performance of the hot-channel analysis in the core thermal-hydraulics.

1. INTRODUCTION

In Pressurized water reactors (PWR) one of the specified safety limits is the departure from nucleate boiling ratio (DNBR), which is a ratio between the heat flux at which the departure from nucleate boiling (DNB) occurs, so-called critical heat flux (CHF), and the actual local heat flux. An accurate prediction of the DNBR is essential in the analysis of a core thermal hydraulics (TH). However, pin wise modelling of a whole PWR core remains the challenging task, since it requires significant computational resources. A lot of efforts and attempts have been made in this direction so far. The current practice is to apply so-called hot channel approach, which consists of two steps. At the first step an assembly-wise analysis is performed, i.e. when all parameters are averaged over fuel assembly (FA) in a core. It allows selecting the "hottest" FA with the lowest DNBR. At the second step, the more detailed pin-wise analysis is made only for the selected FA. The selection of the FA for the pin-wise consideration, the so-called hot assembly, can be made based on several factors, e.g. DNBR, FA power and coolant temperature. This selection judgement of

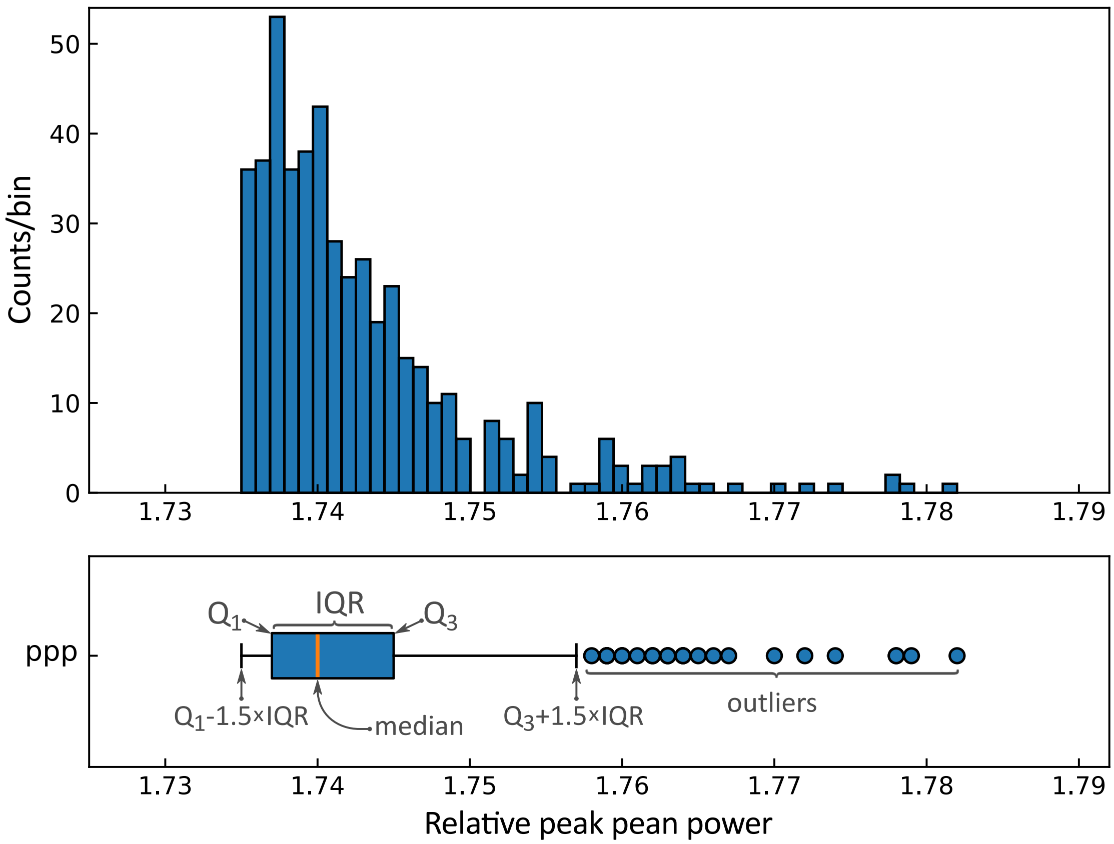

Figure 1 – At the top: a probability density function for the peak pin power (ppp) calculated varying the $^{238}U(n,inl)$ cross section for a realistic PWR core loading. At the bottom: a box plot of the ppp distribution, where IQR is the interquartile range, Q_3 is the 75th percentiles and Q_1 is the 25th percentiles.

the hot channel is also based on the core TH analysis, which requires core power distributions. Such distributions are calculated with core neutronics codes using uncertain neutron cross sections which can affect pin power distribution, and consequently TH parameters. Therefore, in this work, we show how nuclear data uncertainties are affecting the DNBR.

For this purpose, we applied a complex non-normal distribution for the peak pin power (ppp) based on variations of the inelastic scattering cross section of ^{238}U [1]. From the nuclear data point of view, this cross section was expected to modify the ppp in the core, as the fast neutron spectrum and the neutron leakage are heavily dependent on (inelastic) scattering. The ppp distribution was obtained based on the covariance matrix for the $^{238}U(n,inl)$ cross section, as included in the ENDF/B-VII.0 US nuclear data library [2], with the following approach. A simple Monte Carlo process was applied: repeating n times the same neutronics core calculations, each time with a different realization of the $^{238}U(n,inl)$ cross section in agreement with its covariance matrix. As the covariance matrices do not indicate which probability density function to use, a simple normal distribution is assumed. The one-group cross section uncertainty for $^{238}U(n,inl)$ from threshold to 5 MeV is about 20%. It should be noted that other nuclear data uncertainties are not included in this work (e.g. fission cross sections).

Such changes of nuclear data were performed over a series of PWR reactor cycles, taking into account realistic information for the core fuel loading, irradiation history, and shutdown periods and so on. These simulations were performed with the nodal and core simulators CASMO5 and SIMULATE3 using plant information for each cycle. The variations of the cross sections were performed with the SHARK-X tool [3], developed at Paul Scherrer Institut (PSI). The ppp distribution used in this work corresponds to the one obtained at the start of the 6th cycle. The strong observed skewness indicates the non-linear effect of the $^{238}U(n,inl)$ cross section on the ppp. Its origin is described in [1] and comes from a combination of two effects: (1) that the ppp can spatially move depending on the value of the $^{238}U(n,inl)$ cross section, and (2) the effect of the $^{238}U(n,inl)$ cross section is changing if the ppp appears at the center of the core or closer to its border (at the center, a higher $^{238}U(n,inl)$ cross section will decrease the ppp, whereas at the border, a higher $^{238}U(n,inl)$ cross section will increase the ppp).

The main idea of this study is to show how a normal PDF (at the nuclear data level) propagate to a strongly skewed PDF at the ppp (previously obtained in [1]) and how it will effect a TH parameter, in our case, the DNBR parameter. Thus, this study is considered as a follow-up work performed in [1].

Probability density function (PDF) of the ppp for the 481 sampled $^{238}U(n,inl)$ cross section is presented at the top of Figure 1. A common practice in descriptive statistics for the graphical characterization and comparison of PDFs is to use a box plot [4]. Box plots provide basic information about data distribution such as skewness, variability, average and median, quartiles, the lowest and highest data points, and data spread. A box plot, which characterizes the ppp values distribution, is presented at the bottom in Figure 1.

2. FUEL ASSEMBLY MODEL

As a test case, a PWR fuel assembly (FA) with 15×15 rods is selected. We assumed that this FA is operating at nominal conditions, typical for PWR core, which have been calculated with the neutronic code SIMULATE3. Relative axial and radial power distributions are plotted in Figure 2. The selected FA contains 20 guide rods, 205 fuel rods with UO_2 fuel material properties and 6 grid spacers. The subchannel method is applied for modelling this FA, i.e. regions between rods sub-divided into subchannel, such that the power of each rod is split into four surrounded subchannels. The main assumption in subchannel method is that there is only one axial preferential flow direction and subchannels can interact with each other mainly due to pressure difference allowing the transfer of mass, energy and axial momentum between them. We applied the subchannel code COBRA-TF (CTF) [5, 6] to obtain all necessary thermal hydraulic characteristics. This subchannel model consists of 256 subchannels and uniformly distributed 35 axial nodes. The relative axial power distribution is a typical one for PWR type of FA in a middle of a cycle, and each point in the Figure 2(right) is defined for the corresponding axial node. The grid spacer effect is taken into account as an additional pressure drop at the corresponding elevations.

In order to show the effect of nuclear data uncertainty on the DNBR we selected a rod with the minimum DNBR value in FA. The ppp PDF distribution is applied to the variation of the relative radial rod power coefficient with 481 samples. This variation of relative rod power redistributes the fixed total FA power among rods, which is leading to a variation of DNBR value for this rod, as presented in this work. An assessment of three different CHF correlations is performed: W3, Bowring and Biasi [7], and Groeneveld CHF look-up table (LUT) [8] available in CTF.

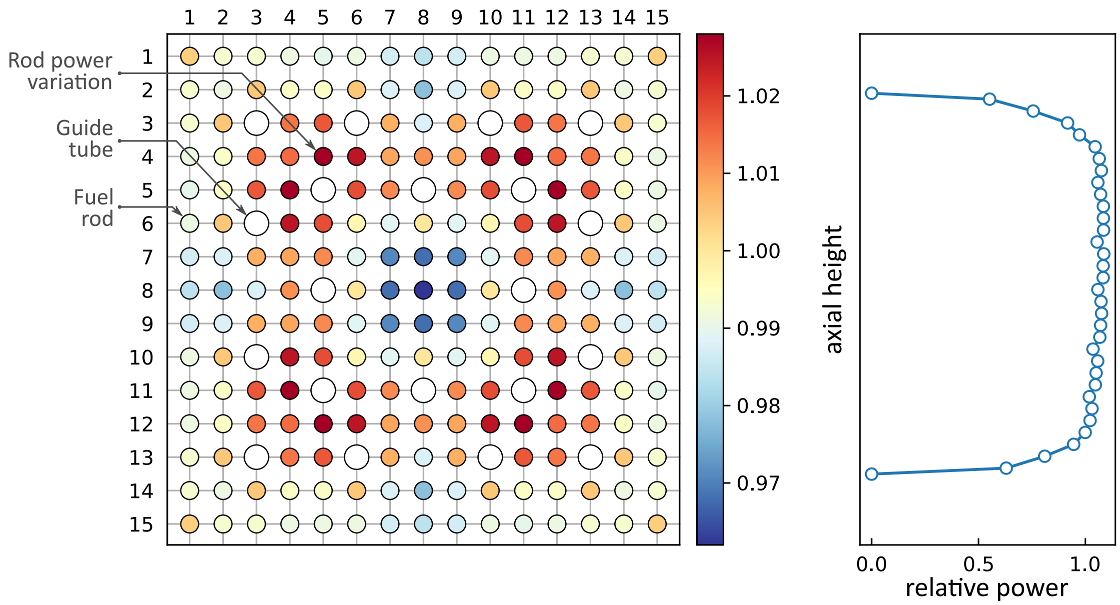

Figure 2 – On the left: a radial relative pin power distribution for the selected fuel assembly of 15×15 rods. Grey lines are representing subchannel subdivision in the fuel assembly. The white colour of the rods is representing a pin with zero power. On the right: an axial relative power profile. Each point is for the centre of a corresponding node.

3. RESULTS

The main results of the DNBR distributions for each CHF correlation are collected in Table 1. Each CHF correlation gives its own range of DNBR values and average values. The difference between DNBR values for CHF correlations is much more significant than the spread represented by the standard deviations (stdv), i.e. the selection of a CHF correlation is the major factor in such DNBR analysis. The fact that at high pressure and mass flux conditions the Biasi correlation is substantially larger than other considered CHF correlations is also noticed in [7]. It is known that TH uncertainties themselves in TH closure models are larger compared to the effect of the nuclear data uncertainty. This is shown, for instance, in [9], where the authors performed uncertainty quantification to boundary conditions and also to selected TH parameters in closure models in subchannel code COBRA-TF. The main outcome and common result for all CHF correlations is that the relative stdv is increased around two times compared to the initial ppp distribution, see Table 1. The measure of the distribution symmetry, i.e. skewness of the initial ppp distribution becomes smaller for DNBR results. The sign change in the skewness from the positive to negative, means that there is a weight change from the right tail of the distribution to the left one. Obtained DNBR stdvs and skewness for all CHF correlations are close to each other. A box plot representation of the results is showed in Figure 3.

4. CONCLUSIONS

The hot channel approach plays a major role in the PWR reactor core safety assessment. It requires accurate judgements for selecting FA and hot subchannel. In this study we showed how nuclear

Table 1 – Comparison between ppp and different CHF correlations for the DNBR average (avg), DNBR standard deviation (stdv), DNBR median (med) and DNBR median absolute deviation (mad).

	ppp	CHF correlations			
		Groeneveld	W3	Bowring	Biasi
avg:	1.7428	2.7595	2.5842	2.8107	3.9375
stdv:	0.0078	0.0218	0.0203	0.0223	0.0291
stdv/avg, %	0.4476	0.7900	0.7855	0.7934	0.7390
med:	1.7400	2.7674	2.5915	2.8188	3.9480
mad:	0.0044	0.0128	0.0119	0.0130	0.0171
mad/med, %	0.2529	0.4625	0.4592	0.4612	0.4331
skewness	1.945	-1.88025	-1.88043	-1.88036	-1.88038

data uncertainties are propagating directly to the TH in the case of the $^{238}U(n,inl)$ reaction and how it may affect the DNBR. This analysis is done for a single burnup step of the selected fuel assembly. Therefore, this work has to be considered as an initial step in such analysis. In future, the same analysis can be repeated, but taking into account nuclear data uncertainties for various reactions, such as elastic, fission or capture, and for all possible isotopes. Here we selected the most important part in nuclear data uncertainty, i.e. inelastic scattering, according to [1]. Also, as the peak pin power distribution varies for different burn-up steps, such study can also be repeated for different cycles and different burn-up values for each cycle. At this step we showed how a normal PDF (at the nuclear data level) propagate to a strongly skewed ppp PDF, and then propagate to a skewed distribution at the TH level. The main result obtained in this work is that nuclear data uncertainty is increasing around two times the range of DNBR distribution, which can be important in the PWR core hot channel methodology. Since currently, all TH codes are deterministic, i.e. after a model preparation and selection of physical models a result is fixed. However, there is another type of uncertainty that can come from the nuclear data, which can affect quite a lot the TH results and as a consequence to the selection of the hottest subchannel.

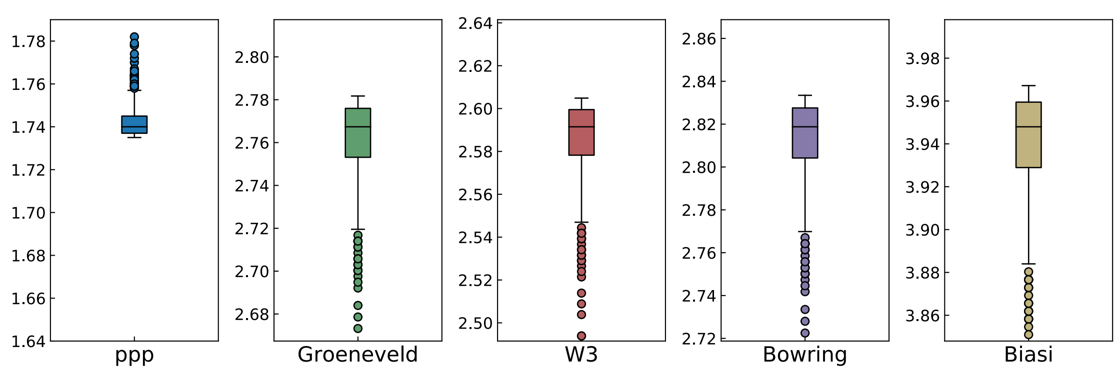

Figure 3 – Box plots comparison between initial ppp distribution and DNBR distributions obtained for different CHF correlations (scale is the same in all figures).

5. ACKNOWLEDGEMENT

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