UNCERTAINTY ANALYSIS FOR LIFT-FORCES ON BWR FUEL ASSEMBLIES

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ABSTRACT

We describe a typical best-estimate model for the lift force acting on the fuel assemblies in a BWR core and discuss how the uncertainty of the model can be assessed and quantified. Accurate modelling of the lift forces and quantification of the uncertainties is essential for the operation of a BWR as the assemblies are held down only by their own weight and hence are able to lift from the core support plate. Would an assembly lift during operation, it could result in damage to core. The issue has attracted some attention lately, as the thermal power and coolant flow rate of many BWRs have been increased substantially, reducing the lift-off margins. Based on the uncertainty analysis, we conclude that a 20-30% margin is recommendable on the calculated lift forces, but these numbers should be verified for every reactor design, as they depend on details in the construction of the reactor and fuel as well as the operating conditions.

KEYWORDS
Lift forces, uncertainty analysis, BWR

1. INTRODUCTION

In light water reactors the fuel assemblies are subject to considerable upwards directed forces due to the coolant water being pumped upwards through the core. These forces, which are of the same magnitude as the weight of the fuel assembly, must be controlled in order to prevent fuel assemblies from lifting from the core support plate. This applies in particular to Boiling Water Reactors (BWR) where the fuel assemblies stand freely on the core support plate, held down only by their own weight, in contrast to a Pressurized Water Reactor (PWR), where the fuel assemblies are equipped with a spring that holds them in place. The present paper covers only the BWR case but many of the methods discussed could, with necessary adjustment, be applied also to the PWR design.

When operating a BWR it is necessary to limit the coolant flow rate (or more precisely the assembly pressure drop) so that the assembly will under no circumstances lift from the core support plate. Would an assembly lift, the consequences could be severe. If the assembly were shifted out of place, it could, in the worst case, block the insertion of a control rod but even a minor lift by a couple of millimeters would let coolant water bypass the fuel channel, leading to inferior cooling of the fuel and possibly dryout and resulting fuel damage. Flow instabilities and mechanical damage due to vibrations are other likely consequences. It should be noted that a lifting fuel assembly could go undetected by reactor operators and automatic safety systems until severely damaged. It is hence necessary to rely on simulations of the lift force, which must be either strictly conservative or very accurate with well quantified uncertainties (the best estimate approach).

Traditionally, the lift force issue has not attracted very much attention as the margins to lift-off have been large and other concerns have been more limiting. However, many BWRs have gone through significant power up-rates and have in many cases been equipped with new and more powerful circulation pumps. As a result, the lift force margins have decreased to the point where they may be limiting for the reactor operation. It is hence of interest to study the lift force models in some depth and, possibly, to replace overly conservative assumptions with best estimate models.

The present paper discusses the physical principles behind the lift force models that are in use by the nuclear industry today, the uncertainties inherent in these models and how uncertainty propagation and statistical techniques can be utilized to quantify the lift-off margins. Finally the issues of operating margins and acceptable failure risk are shortly discussed. More details about the modelling and simulations presented in the present paper have been published in Swedish [1].

2. LIFT FORCE MODELS

The lift force on a fuel assembly is the sum of all the vertical hydraulic forces (static and dynamic) acting on every surface of the assembly. As the geometry of the assembly is very complex and subject to a highly turbulent two-phase flow, the lift-force problem may seem to be a difficult one. It should, however, be noted that the pressure drop over the fuel assembly can be calculated relatively accurately be means of empirical correlations. This fact can be exploited to create a relatively simple lift-force model. The trick is to enclose the fuel assembly in a control volume of simple shape, e.g. a vertical rectangular block that closely follows the outer boundary of the fuel channel (Fig. 1). Assuming that the pressure drop over the fuel assembly is known, the static pressure on every point on the outer boundary of the control volume is also known and can be easily integrated into a resultant force and – because the control volume does not move – the sum of all forces acting on it must be zero. Therefore, the lift force, which is precisely the force from the control volume on the fuel assembly, can be calculated by a simple force balance.

Some more complications must be added to achieve a complete model, however. One has to account for the acceleration of the coolant (due to boiling) through the fuel channel and the pressure loss over the core support plate. In the ABB BWR design it is also possible to lift the fuel bundle (rods and spacer grids) within the fuel channel, without lifting the entire assembly from the core support place. There are hence two different lift forces to consider: the assembly lift acting on the entire fuel channel and inlet piece and the bundle lift force acting on bundle inside the fuel channel, respectively. (Due to different design this issue may not present in all BWRs). A slightly simplified description of these two lift force models follow in the next section.

It is common to perform lift-force calculations for one or several "hot-channels" in the core with conservative assumptions but there is today nothing preventing that every assembly in the core is studied throughout the cycle with best-estimate (i.e. realistic) boundary conditions. Then, however, it is of great importance to consider the uncertainties inherent in both model and boundary conditions and that is the primary motivation for the present work.

2.1. Bundle lift force

The force balance on the fuel bundle can be expressed as follows:

$$F_{\Delta P} + F_{acc.} - F_{fluid\,mass} - F_{bound.fric.} - F_{lift} = 0 \tag{1}$$

where the terms represent, in order from left to right, the force acting on the control volume due to the pressure gradient, fluid acceleration, fluid mass weight, boundary friction terms and, finally, the assembly lift force (or, strictly speaking, its reaction force on the control volume). Similar models have been discussed by e.g. [2]. Note that if the lift force becomes greater than the assembly weight, the assembly will lift. When the lift force is less than the assembly weight the assembly would sink if it did not rest on the core support plate. The support force from the core support plate on the fuel assembly is not included in eq. (1) because it does not at on the control volume (only on the fuel assembly). Note also that eq. (1) is valid only for steady-state; transients are beyond the scope of the current work.

The first term in eq.(1) is calculated as

$$F_{\Delta P} = \Delta P_{CV} A_{CV} \tag{2}$$

where ΔP_{CV} and A_{CV} denote the pressure drop over the control volume and the cross-section area of the control volume, respectively. When the coolant is boiling, the density at the outlet is significantly less than the density at the inlet and the velocity consequently higher, creating a negative (i.e. hold-down) force on the assembly. This acceleration force term becomes

$$F_{acc.} = \dot{m}^2 \left(\frac{1}{\rho_{out}\, A_{out}} - \frac{1}{\rho_{in} A_{in}} \right) \tag{3}$$

Another force, which gives a hold-down contribution is due to friction with the channel box wall. It is relevant only when analyzing the force on the bundle as a free component inside the fuel channel. The pressure loss, as calculated by correlations, includes both the friction with the fuel rods and the channel box wall. The latter contribution should be removed as it does not contribute to the lift on the fuel bundle (Fig. 2). Assuming that the frictional shear force is the same at all wetted surfaces, this can be written as

$$F_{bound.fric.} = F_{\Delta P,fric} \frac{U_{box}}{U_{tot}} \tag{4}$$

where U_{box} is the perimeter of the channel box wall and U_{tot} is the total wetted perimeter. Finally the gravitational force acting on the fluid inside the control volume should be subtracted, according to

$$F_{fluid\,mass} = [\bar{\alpha}\rho_v + (1 - \bar{\alpha})\rho_l] \left[V_{CV} - V_{fuel}\right] g \tag{5}$$

where $\bar{\alpha}$ is the bundle average void fraction, $\rho_v$and ρ_l are the steam and liquid densities, respectively, V_{CV} is the entire control volume and V_{fuel} is the part occupied by the fuel and box material. Finally, g is the acceleration of gravity.

It is interesting to note that in the special case when there is no coolant flow through the assembly ($\dot{m} = 0$), the only pressure drop is due to gravity and the lift force simplifies to the classical Archimedes expression for buoyancy:

$$F_{lift} = V_{fuel} \rho_l g \tag{6}$$

assuming that the assembly is filled with liquid water. This example illustrates that the buoyancy force is implicit in equation (1) and should not be added separately.

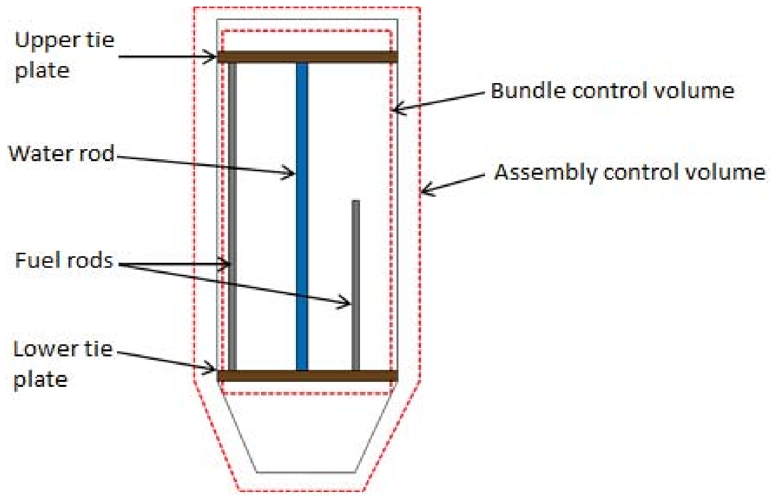

Figure 1. Simplified sketch of typical BWR fuel assembly and the control volumes used in the derivation of the lift force equations.

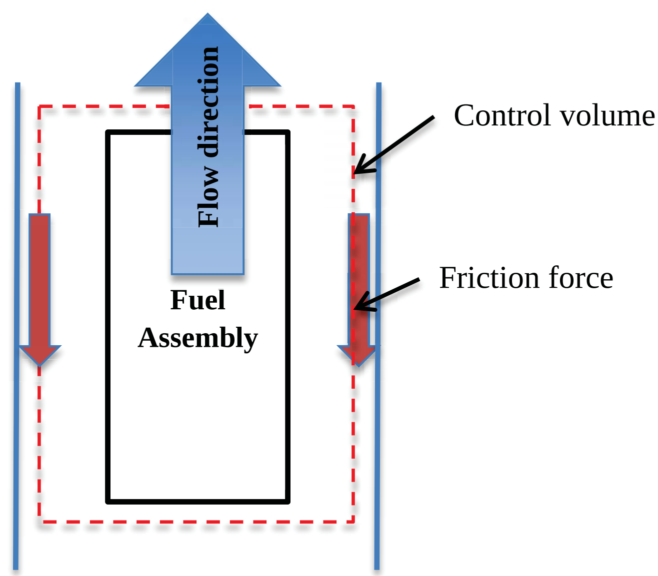

Figure 2. Detailed view of the inner control volume, illustrating the box wall friction force acting downwards on the fluid inside the control volume.

2.2 Assembly lift force

The equations for the entire assembly (including the fuel channel) are very similar to the ones used for the bundle but there are some differences. First, the control volume extends further down, including the inlet piece but not the orifice in the core support plate (Fig. 1). The pressure drop is therefore significantly larger. Second, the negative wall friction term does not apply when the entire assembly is analyzed, as there is no wall at the outer boundary of the control volume. Instead, as small (insignificant) positive lift force from the bypass flow applies to the outside of the fuel channel. Interestingly, the assembly lift force is still typically smaller than the one acting on the bundle alone (at least relative to weight of the assembly and bundle, respectively). The reason is that the part of the control volume that is exposed to the high pressure below the inlet piece has a relatively small area. The remaining part of the lower boundary is exposed to the much lower pressure in the bypass region of the core, as illustrated in Fig. 3. This means that the bundle lift force, when it applies, is typically the limiting one in the analysis.

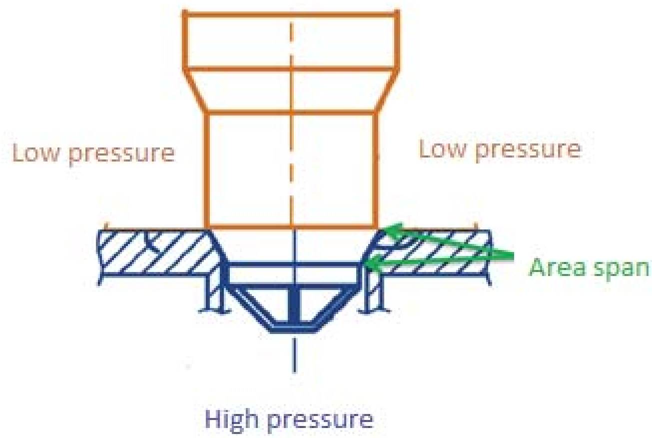

Figure 3. The inlet piece of the fuel assembly, standing on the core support plate.

3. A SIMPLE PIRT FOR LIFT FORCE CALCULATIONS

A simplified Phenomena Identification and Ranking Table (PIRT) was developed as a guidance when selecting probability distributions and approximations. The PIRT was based on analytical studies of the lift force equations, numerical experiments with the core simulator POLCA7 [3] as well as expert judgment in relation to approximations and assumptions inherent in the lift force model and thermal hydraulic modelling in the core simulator. The results is summarized in Table 1. This PIRT should be viewed as essentially a sensitivity study with the POLCA7 core simulator combined with the authors' understanding of the physical process involved.

Table 1. Simplified PIRT for lift force calculations

Reactor parameters	Rank	Assembly parameters	Rank
Core flow	High	Relativ assembly power	Medium
Reactor power	Medium	Two-phase multiplier	High
System pressure	Low	Channel flow	High
Inlet temperature	Low	Assembly pressure drop	High
Core pressure drop	High	Assembly void fraction	Low
Average void fraction	Low	Water rod void fraction	Low
		Leakage to bypass	Low
		Coolant acceleration	Medium
		Inlet pressure loss	High
		Cross-section area	High

4. SELECTION OF PROBABILITY DISTRIBUTIONS

The selection of probability distributions was based loosely on the idea of the maximum entropy principle [4]. Thus, in cases where a definite upper and lower bound can be determined for the uncertainty, a uniform distribution has been used (e.g. for mechanical manufacturing tolerances). When mean value and standard deviation have been estimated, a normal distribution was used. Note that the use of a normal distribution in these cases were not motivated by any normality test but only by the fact that the maximum entropy distribution when the mean and variance are known is the normal distribution.

4.1. Core pressure drop

The pressure drop calculated by the core simulator is uncertain for two different reasons. First, the state of the reactor is to some extent uncertain. The most important parameters being the coolant flow through the core and the reactor power. Second, there is a significant modelling uncertainty, mostly related to the two-phase pressure loss correlations. The first contribution was estimated to account for about 1.5% uncertainty on the core pressure drop, using known values for uncertainty of core flow and power (e.g. 1.0% and 1.2%). The modelling uncertainty was estimated to 4.3%, based on comparison between various, commonly used correlations for the two-phase multiplier and friction factor. This model-to-model approach was selected in lack of accurate measurements of the core pressure drop. The total uncertainty (standard deviation) in the calculated core pressure drop was thus estimated to $\sqrt{1.5\%^2 + 4.3\%^2} \approx 4.6\%$. This uncertainty was applied to the two-phase pressure loss term (section 4.5 below).

4.2. Channel coolant flow rate

The coolant flow rate in individual assemblies is calculated by the core simulator. The uncertainty (standard deviation) can be estimated based on in-reactor measurements in certain, instrumented measurement positions (at least in the ABB internal pump BWR design). A typical value amounted to about 1.5%, which was applied to the inlet pressure drop (section 4.4 below).

4.3. Assembly power

The assembly power is calculated by the core simulator and hence carries some modelling error. This error is, however, tightly correlated with the error in the calculated assembly coolant flow rate (because the two-phase pressure drop is linked to the channel void fraction and the core flow is redistributed until all channels have the same total pressure drop). Noting that the assembly void fraction, in itself, has only a minor impact on the lift force, the impact of the assembly power uncertainty has already been included in the assembly coolant flow uncertainty. Because it should not be included twice, the assembly power uncertainty was excluded from the present uncertainty analysis.

4.4. Inlet pressure drop

The uncertainty in the inlet pressure drop was based on the uncertainty in the inlet coolant flowrate (section 4.2) and the uncertainty in the measurement of the loss-coefficient, which was small in comparison.

4.5. Two-phase pressure drop

The uncertainty in the two-phase pressure drop was judged to be mainly due to modelling uncertainties (the two-phase multiplier). Other contributions come from the uncertainty in the assembly flow rate and power and measurement uncertainties related to the determination of loss coefficient for spacer grids. Also the uncertainty in the axial power shape could be lumped into the two-phase pressure drop uncertainty but it is relatively small (core simulators predict the power shape quite accurately). These are essentially the same uncertainty sources that were considered in the analysis of the core pressure drop (section 4.1). The combined core pressure drop uncertainty was hence applied to the two-phase pressure drop term in the lift-force model.

4.6. Channel wall friction

The channel wall friction term carries an interesting modelling uncertainty. The assumption that the same friction coefficient applies to all wetted walls inside the assembly, which was used to derive eq. (4) is not obvious. In particular, it should not be assumed that the frictional shear stress on the fuel channel box wall is the same as on the fuel rods. It is, however, not easy to quantify this uncertainty as no relevant measurements exist. One could possibly use CFD simulations as a reference but even these are somewhat dubious in two-phase boiling flows. In the present work we have made minimal assumptions: the friction on the box cannot be less than zero and we have, conservatively, assumed that it is not larger than the corresponding shear on the fuel rods. Thus having defined the upper and lower bounds, we use a uniform probability distribution, as stipulated by the maximum entropy principle.

4.7. Inlet piece exposed area

The outside of the inlet piece is exposed partly to the high pressure under the core support plate and partly to the lower pressure in the bypass region. Because the contact area between the inlet piece and the core support plate is sloped, it is not obvious where the boundary between the higher and lower pressure is located (Fig. 3). This has therefore been treated as an uncertainty in the model. In this case the known geometry of the inlet piece yields an upper and a lower bound for the uncertainty and a uniform distribution was hence used also here.

5. UNCERTAINTY PROPAGATION

5.1. Brute force Monte Carlo

The uncertainties were essentially propagated through the core simulator and lift force model by brute force Monte-Carlo. However, in order to reduce the computational time, parameters that occur only in the lift force model but not in other models of the core simulator were propagated by running the lift force model as a separate post-processing step. The parameters that relate to the assembly pressure drop were also propagated using this method, even though they affect the core conditions. The core-wide effect is, however, negligible as long as those variable are varied randomly and independently.

The uncertainty propagation results in a probability distribution for the lift force and the probability of assembly or bundle lift-off can be calculated simply by counting the number trials where the lift force exceeds the physical weight of the component as illustrated in Fig. 4.

A confidence factor should also be applied to account for the probabilistic method of uncertainty analysis. We calculated this factor on the 95% confidence level based on the order statistics and formula for the tolerance interval presented in [5]. Due to the very large number of trials (typically 100 000) the confidence factor was negligibly small but has nevertheless been included in the analysis.

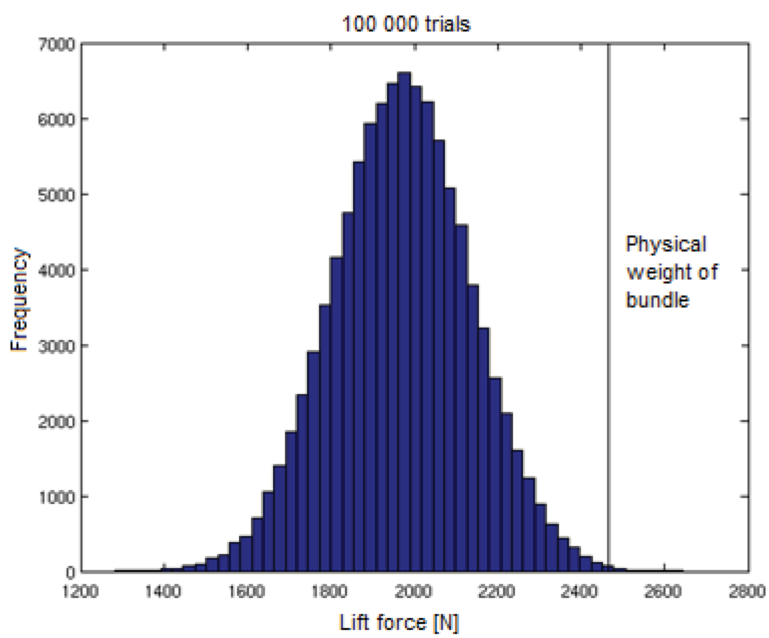

Figure 4: Example of probability distribution for bundle lift force generated by 100 000 Monte Carlo trials. Lift probability is calculated by counting the number of trials exceeding the physical weight of the bundle.

5.2. Wilks method

The Wilks method [6] is commonly used to calculate an upper bound to an arbitrary distribution, with a specified coverage and confidence level. The method can significantly reduce the computational cost associated with Monte Carlo propagation of uncertainties (down to 59 trials for a one-sided 95/95 coverage). It is, however, mostly useful to verify that there is (more than) sufficient margin to some given safety criterion. If the margin to the criterion is small, or if the purpose of the analysis is to define the criterion, the variability in the results from the Wilks method can be problematic. As our main goal was to define a safety factor based on the results from the uncertainty propagation and the standard (first order) application of Wilks' formula showed unacceptable variation, we based all conclusions on the brute force Monte Carlo simulations. We were, however, able to achieve reasonably robust results by using Wilks formula at order 6 or 8. These results were, as expected, slightly more conservative than the once obtained using a much larger number of trials. A numerical example is shown in Table 2; the spans presented for the Wilks methods represent the variability encountered when repeating the analysis several times (e.g. by running several sets of 59 trials and comparing the bounding value from each set).

Table 2: Examples of calculated upper bounds to the bundle lift force with different methods.

Coverage	Confidence level	Wilks order 1 (59 trials)	Wilks order 4 (153 trials)	Wilks orer 8 (260 trials)	Monte Carlo (100 000 trials)
95 %	95 %	2250-2470 [N]	2240-2410 [N]	2240-2340 [N]	2260 [N]
99 %	95 %	2380-2600 [N]	2370-2450 [N]	2390-2420 [N]	2380 [N]
99,9 %	95 %	2510-2670 [N]	2480-2560 [N]	2500-2550 [N]	2500 [N]

6. RESULTS

Using actual Swedish BWRs with realistic operating conditions as examples we have found that lift force uncertainties depend to some extent on the operating conditions and fuel design. The most limiting operating conditions is always the highest allowable reactor power and coolant flow rate and the limiting assembly in the core will be the one with highest relative power and bottom shifted axial power distribution (to maximize the two-phase pressure loss). It is always recommendable to select a conservative hot bundle power and axial power distribution in order to cover operational variations.

The exact numbers relating to any particular reactor and fuel design cannot be disclosed here but in general the results were as follows: For 95% probability of non-lift, a 15% margin must be added to the lift force calculated by the best-estimate model in order to cover the uncertainties. For a 99% probability, the margin should be around 20% and in order to achieve 99.9% probability the necessary margin approaches 30%. All these numbers were calculated on a 95% confidence level (with a very large number of Mont-Carlo trials and hence a very small penalty for the statistical error).

7. DISCUSSION AND CONCLUSIONS

One potentially important aspect has not been treated in the present work: The analysis has dealt only with the most limiting (highest power) assembly in the core, assuming that it bounds all other assemblies. However, when applying uncertainties it is possible that another assembly becomes the limiting even if it is nominally not so. In other words, the more assemblies there are in the core, the higher the probability that some of them would lift. This issue is far from trivial to assess. Not only must every assembly in the core be analyzed according to its own state (power etc.) but one has to decide whether uncertainties should be applied independently to each assembly in the core or if they are correlated (statistically dependent). The degree of statistical independence should most likely depend on the type of uncertainty considered. For example, the reactor power uncertainty would affect all assemblies in the same way while the assembly power uncertainty affects each assembly independently. For other uncertainties, the choice between dependent and independent may be less obvious. It is, nevertheless, highly significant. Applying uncertainties independently will greatly increase the resulting lift probability as compared to applying the same uncertainties in a correlated manner (the former is hence the conservative approach).

We have also not discussed accident scenarios. The entire analysis presented here was carried out for normal, steady operating conditions (even though the most limiting conditions were used, i.e. highest allowable power and flow rate). Fortunately most accident scenarios would result in reduced lift forces (because SCRAM or pump run-down decrease reactor power and coolant flow rate). Any transient increasing the power significantly above the 100% level, e.g. a failure in the power controller, would quickly initiate reactor SCRAM. In relation to lift forces the most severe accident scenario would probably be a steam line break, where, during the initial blow-down and flashing phase, the core pressure drop could increase significantly. This is, however, a LOCA-scenario and some degree of core damage can be accepted (but not the dislocation of an entire fuel assembly).

Based on the results of the current study, we suggest that at least a 30% lift force margin, corresponding to 99.9% probability, be applied in cases when a lift-off incident could affect the reactor safety (e.g. blocking a control rod). In less critical situations, when lift-off would cause only minor fuel damage (e.g. lifting a bundle within the fuel channel), a 20 % margin, corresponding to 99% probability could be sufficient. Note, however, that these numbers are only examples and depend on the design of the reactor core and the fuel as well as the operating conditions of the reactor and the accuracy of the modelling in the core simulator. They should hence not be applied without going through the analysis described here, using data describing the analyzed reactor and fuel.

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